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13.0 CONDUCT OF OPERATIONS

The South Texas Project Electric Generating Station (STPEGS) is currently owned by NRG South Texas LP, City Public Service Board of San Antonio (CPS), and City of Austin (COA) as tenants in common. Houston Lighting & Power Company (HL&P) was the project manager of the STPEGS Units 1 and 2, and as such was responsible for the engineering, design, licensing, construction, startup, and initial operation of the STPEGS units.

HISTORICAL INFORMATION

Brown & Root (B&R) initially performed the engineering, design, procurement, and construction activities for STPEGS. In September 1981, Bechtel Energy Corporation (BEC) began preparations for assuming responsibility relative to engineering, design, procurement, and construction management. During the transition period, HL&P coordinated the activities of B&R and BEC to effect an orderly transfer of engineering, design, procurement, and construction management responsibilities to BEC. After transition, BEC became responsible for engineering, design, procurement, and construction management. Ebasco Services, Inc. became responsible for construction.

On November 17, 1997**,** responsibility for operating the South Texas Project was transferred from HL&P to the STP Nuclear Operating Company (STPNOC). STPNOC is a Texas non-profit corporation created, controlled and financed by the Owners specifically for the purpose of operating STP. This change was incorporated in the Unit 1 and Unit 2 Operating Licenses via Amendments 93 and 80, respectively.

On August 31, 2002, HL&P (Reliant Energy) transferred its ownership interest in STPEGS and in STPNOC to Texas Genco, LP. This change was incorporated in the Unit 1 and Unit 2 Operating Licenses via Amendments 142 and 130, respectively.

On January 12, 2006, the NRC approved the transfer of ownership interest in STPEGS and in STP Nuclear Operating Company from Texas Genco, LP to NRG South Texas LP. This change was incorporated in the Unit 1 and Unit 2 Operating Licenses via Amendments 178 and 165, respectively.

American Electric Power Company, Inc. (AEP) is the parent company of CPL. Effective December 23, 2002, AEP sold the retail company portion of CPL to Centrica, and transferred the Central Power and Light/CPL names to Centrica as well. AEP renamed the remaining portions of CPL (generation and distribution interests) as “AEP Texas Central Company” (TCC). TCC sold its STP ownership interest to CPS and Texas Genco, LP on May 19, 2005.

AEP Texas Central Company sold its ownership interest in STPEGS to Texas Genco, LP and CPS. License Amendments 172/160 became effective on May 19, 2005. Texas Genco, LP sold its STP ownership interest to NRG South Texas LP. License Amendments 178 and 165 became effective on June 29, 2007.

Westinghouse Electric Corporation (Westinghouse) designed, fabricated, and delivered the Nuclear Steam Supply System (NSSS), turbine generator (TG) unit and its auxiliaries. Westinghouse also designed, fabricated, and delivered the nuclear fuel assemblies for the initial core loading.

HISTORICAL INFORMATION

13.1 ORGANIZATIONAL STRUCTURE

13.1.1 Management and Technical Support Organization

13.1.1.1 Design and Operating Responsibilities. The following paragraphs summarize how the design, construction, and preoperational activities were accomplished, and describe specific responsibilities and activities relative to technical support for operations.

13.1.1.1.1 Design and Construction Activities (Project Phase):

13.1.1.1.1.1 Principal Site-Related Engineering Work –

1. Meteorology

A preoperational meteorological monitoring program was established at the site in July 1973 to provide those meteorological factors that bear upon plant design, operation, and safety. The program
was conducted by NUS Corporation until July 1975, with the responsibility for the program being assumed by the HL&P (historical context) Environmental Air Quality Division at that time. Data collected through September 1977 was used for design and licensing purposes. The monitoring system continued to collect data until 1982, at which time it was shut down for upgrading in accordance with current requirements.

2. Geology and Seismology

The geological and seismological investigations and evaluations were conducted by Woodward-Clyde Consultants (WCC). The geotechnical engineering investigations and analysis for the plant location and Essential Cooling Pond (ECP) were also conducted by WCC. A major portion of the field boring and sampling program, and some specialized laboratory and field studies associated with this work were done by subcontractors under the supervision of WCC personnel. The geotechnical investigations and analysis for the reservoir and reservoir-related facilities were conducted by McClelland Engineers, Inc.

Design and evaluation activities in the plant and reservoir areas in connection with earthwork and foundation construction were conducted by B&R. WCC conducted the foundation verification and geological mapping activities associated with Category I plant facilities, and provided continued consultant services to B&R for plant area geotechnical design-related issues during construction. Pittsburgh Testing Laboratory provided quality control inspections and laboratory testing for all plant and ECP area earthwork during construction. McClelland Engineers, Inc. provided consultant services and laboratory testing for the reservoir-related earthwork during construction. Overall direction of all geological, seismological, and geotechnical engineering studies was by B&R, HL&P, and/or BEC.

3. Hydrology

B&R initially developed the probable maximum flood from offsite areas for STPEGS based upon hydrologic investigations of the Colorado River Basin previously made by the Fort Worth, Texas,

District Office of the U.S. Army Corps of Engineers (USACE). Physical data, previous reports, and unpublished engineering studies, together with technical guidance, were made available by both the
Fort Worth and Galveston District USACE offices. Detailed information concerning hydrology is given in Section 2.4.

4. Demography

B&R performed the initial demographic studies relative to population distribution near the plant. HL&P (historical context) has since updated these population distributions. See Section 2.1.3 for details.

5. Environmental Effects

A preoperational monitoring program for STPEGS was developed to enable the collection of data necessary to determine possible impacts on the environment due to construction activities and to establish a preoperational baseline from which to evaluate future environmental monitoring. This program is described in the STPEGS Environmental Report (ER).

An Environmental Protection Control Plan provided for periodic review of all construction activities to ensure that those activities conformed to the environmental conditions set forth in the construction permit. No harmful effects or evidence of irreversible damage were detected by the monitoring program during the construction period.

Preoperational radiological monitoring activities were conducted.

13.1.1.1.1.2 Design of Plant and Ancillary Systems - Design and construction of the plant and its ancillary systems are complete.

13.1.1.1.1.3 Review of Plant Design Features - Design control and review is performed in accordance with the Quality Assurance Program for STPEGS Units 1 and 2, as discussed in the Operations Quality Assurance Plan (OQAP).

13.1.1.1.1.4 Development of Safety Analysis Report - Overall responsibility for update of the UFSAR rests with STPNOC Licensing.

13.1.1.1.1.5 Review of Material and Component Specifications - All safety-related project specifications are reviewed in accordance with the OQAP.

13.1.1.1.1.6 Management and Review of Construction Activities - HL&P construction management performed the following management and control activities at the construction site after the change to BEC on December 16, 1982.

1. HL&P construction personnel monitored field activities in order to assure that the STPEGS construction was in compliance with design and accomplished by the most expeditious, economical, and safe means possible. This monitoring included: (1) random inspection of

ongoing construction activities for conformance to specifications, drawings, and procedures, and (2) general review of the security, safety, and environmental programs. Construction management played an active part in the resolution of problems identified during field monitoring. The construction organization provided the action, direction, and coordination required by other groups, internal or external to the Company, for the prompt resolution of these problems.

2. The contractor's cost and schedule performance was monitored to keep construction management and project management informed of project status. The Construction Department evaluated ongoing construction activities for cost and schedule impacts, and actively participated in identifying, selecting, and implementing resolutions to eliminate or reduce the effect of such impacts.

13.1.1.1.2 Preoperational Activities:

13.1.1.1.2.1 Development of Human Engineering Design Objectives and Design Phase Review of Proposed Control Room Layouts - The human engineering concepts and objectives used in the control room design have been provided to the NRC as part of the detailed Control Room Design Review.

13.1.1.1.2.2 Development and Implementation of Staff and Training Programs - The training programs utilized for this facility are described in Section 13.2. This program is in accordance with the schedule indicated in that section.

13.1.1.1.2.3 Development of Plans for the Initial Test Program - The initial plant test program for STPEGS is described in Chapter 14. The information presented in that chapter is supplemented by the STPEGS Startup Manual and the STPEGS Plant Procedures Manual. These documents provided administrative guidance during the initial plant test program, and contain instructions and procedures which address various test activities. These activities included, but were not limited to, organization, interfaces, test procedure development, review and approval, schedules, document control, calibration of test equipment, housekeeping and system cleaning, test conduct, and deficiency resolution.

Nuclear Plant Operations Department (NPOD) procedures for the STPEGS were utilized, as appropriate, by Startup to effect a coordinated transition from the startup phase to the operational phase of the project.

13.1.1.1.2.4 Development of Plant Maintenance Programs - Responsibilities of the resident maintenance forces are described in Section 13.1.2, "Operating Organization."

The STPEGS maintenance program enables equipment to be maintained in a reliable condition and satisfy the requirements of the regulatory agencies having jurisdiction. Maintenance of those structures, systems, and components which prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public is continued in accordance with the requirements of the OQAP.

HISTORICAL INFORMATION

The maintenance staff is sized to perform the routine and preventive maintenance work load. The station staff is supplemented by HL&P (historical context) central maintenance forces or by contract
forces during overhaul and repair of major equipment and systems. Maintenance is performed under the direction of cognizant supervisors in accordance with accepted work practices and procedures.

The scope and frequency of preventive maintenance is based on past experience with similar equipment and the manufacturer's recommendations. Suitable records are kept to establish the maintenance history of major safety-related equipment. Written procedures, orders, and instructions which govern maintenance include sufficient detail to ensure satisfactory completion of the work, but do not necessarily include detailed step-by-step delineations of basic skills normally possessed by qualified maintenance personnel. Maintenance work is preplanned except for emergencies. Routine training meetings are held to foster safety awareness and quality of workmanship.

HISTORICAL INFORMATION

13.1.1.1.3 Technical Support for Operations: The Engineering Department provides technical support in the areas of nuclear safety, engineering, and nuclear fuel, to support testing and operation of STPEGS. The Engineering staff is described in Section 13.1.1.2.2. Additional technical support is provided in the Operating Group as described in Section 13.1.2.

Radiochemistry and health physics support is an integral part of the Nuclear Plant Operations staff.

If technical support cannot be provided by STPNOC, then a consultant will be retained.

13.1.1.2 Organizational Arrangements.

STPNOC is managed by a Board of Directors which consists of four members. Each owner designates one member for the Board. In addition, the President & Chief Executive Officer for STPNOC serves on the Board and reports to the Board.

Ultimate responsibility for design, procurement, construction, testing, quality assurance, and operation of STPEGS rests with the President & Chief Executive Officer. The President & Chief Executive Officer assigns responsibilities to the various STPNOC organizations described below.

13.1.1.2.2 STPNOC Organization:

STP Nuclear Operating Company is comprised of the Generation, Oversight, Engineering, Financial Support, and Station Support Departments. Refer to the OQAP for a description of the organization and responsibilities.

13.1.1.3 Qualifications. Within STPNOC, the person whose job position most closely corresponds to that identified as "engineer-in-charge" as defined by ANSI N18.1-1971 is the President & Chief Executive Officer.

Resumes and training records of key management and engineering personnel are available on site. The resumes of the fire protection engineers who performed the fire hazards analysis are contained in the Fire Hazards Analysis Report.

Qualifications of STPEGS personnel conform to RG 1.8 (refer to Table 3.12-1). The OQAP addresses the qualification, training, and certification of personnel.

13.1.2 Operating Organization

The functions and responsibilities of various positions at STPEGS, including a specific succession to responsibility for overall operation of the plant in the event of absences, incapacitation of personnel, or other emergencies, are described in this section.

Plant General Manager

The Plant General Manager is responsible for the safe, reliable, and efficient startup, operation, maintenance, and refueling of the units, as well as adherence to all requirements of the Operating Licenses and the Technical Specifications. The Generation Department Organization is shown in Figure 13.1-1.

Operation Director

The Operations Director reports to the Plant General Manager. He is responsible for planning overall activities and work of Operations personnel in cooperation with other department heads to develop an integrated plant operations program with the primary objectives of reactor safety and plant reliability. CN-3235

Operations Division Manager – Unit Operations

The Operations Division Manager – Unit Operations reports to the Operations Director and ensures that the units are operated in accordance with plant procedures and operating license requirements. The Unit Operations Division Manager will hold a current SRO license in accordance with Technical Specification 6.2.2. CN-3235 and CN-3239

The Shift Operations organization is shown in Figure 13.1-2.

Shift Manager (SRO)

The Shift Manager reports to the Operations Division Manager- Unit Operations. The Shift Manager for a unit is responsible for directing the activities of the Plant Operations personnel assigned to their shift. The Shift Manager is responsible for assuring that shift operations are performed in accordance with approved procedures, the Operating Licenses, and the Technical Specifications. The Shift Manager authorizes the placement of systems, components, and equipment in or out of service as required for the safe and efficient operation of the plant in accordance with the Technical Specifications. CN-3235 and CN-3239

Unit Supervisor (SRO)

The Unit Supervisor is responsible for supervising the Plant Operations personnel assigned to his unit and for directing control room activities to assure safe and efficient unit operation in accordance with the Operating Licenses, Technical Specifications, and approved procedures during his shift. He is cognizant of all work or tests which may affect the operation of the unit in accordance with administrative control procedures. He directly supervises control room activities during startup, shutdown, abnormal, and emergency conditions. He reports to the Shift Manager of his assigned unit and may assume the duties and responsibilities of the Shift Manager in the event he is unavailable.

Reactor Operator (RO)

The Reactor Operator is responsible for the safe and efficient operation of the control room equipment of his assigned unit in accordance with the Operating Licenses, Technical Specifications,
and approved procedures during his shift. He monitors and controls unit parameters and unit equipment from the control room, and performs required operational checks and surveillance tests. He initiates the immediate actions necessary to maintain the unit in a safe operating condition during abnormal and emergency conditions. He maintains required records, logs, and charts of unit data, shift events, and performance checks. He initiates requests for equipment repairs, and clears and tags equipment as directed by shift supervision. The Reactor Operator reports to the Unit Supervisor of his assigned unit.

Plant Operator

The Plant Operator is responsible for safe operation of systems and equipment as directed from the control room of his assigned unit. The Plant Operator monitors plant parameters as required to be aware of plant conditions, performs required operational checks, initiates requests for equipment repairs, clears and tags equipment as directed, and maintains required logs, charts, and records of plant data, shift events and performance checks on his shift. The Plant Operator reports to the Unit Supervisor of his assigned unit.

Administrative Aide

In accordance with NUREG –0737, item I.A.1.3, administrative functions that detract from the management responsibility for assuring the safe operation of the plant are delegated to other Operations personnel not on duty in the control room. An Administrative Aide has been assigned to perform routine administrative duties and processes such as routing records, logs, and correspondence for the Control Room Operations staff as required.

Manager, Maintenance

The Manager, Maintenance reports to the Plant General Manager. The Manager, Maintenance is responsible for mechanical, electrical, instrument and control (I&C), and support activities. Responsibilities consist of ensuring that mechanical, electrical, and I&C (including direction for proper installation, calibration, testing and maintenance) systems of all plant facilities are maintained to assure their dependability, reliability and operating efficiency to comply with the requirements of the Operating License and the Technical Specifications. The Manager, Maintenance is also responsible for corrective and preventive maintenance for both units and common support facilities of the plant.

13.1.2.5 Operating Shift Crews. The minimum operating shift crew for STPEGS is listed in the Technical Specifications.

In addition to the operating shift crew, a Radiation Protection Technician will be onsite at all times when fuel is in either reactor to ensure that adequate radiation protection coverage is provided for station personnel. The Radiation Protection Technician will inform the Shift Manager of plant radiological conditions.

A site Fire Brigade of at least five personnel who may have normal shift duties, but are trained specifically in fire protection, is maintained on site.

13.1.3 Qualifications of Nuclear Plant Personnel

Key personnel assigned to STPEGS have had extensive experience in steam electric stations in their respective areas of responsibility, and they will be given nuclear training where necessary to prepare them for their specific assignments at the plant. Section 13.2 discusses the nuclear training program for these personnel.

13.1.3.1 Qualification Requirements. The qualification requirements for plant supervisory, operating, technical, and maintenance support personnel at STPEGS meet or exceed the guidance given on personnel qualifications contained in RG 1.8 with clarification provided in Table 3.12-1. Plant operating personnel meet the experience requirements of Generic Letter 84-16.

13.1.3.2 Qualifications of Plant Personnel. Resumes of managerial and supervisory personnel are maintained on file at the plant.

13.2 TRAINING PROGRAM

The goal of the Training Programs is to provide qualified personnel to operate and maintain the South Texas Project Electric Generating Station (STPEGS) in compliance with license requirements, Technical Specifications, and appropriate governmental regulations. The Training Programs are designed using a Systematic Approach to Training concept and provide the required training based on individual employee experience, the intended position, and previous training/education. The Training Programs provide reasonable assurance that fully trained and qualified operating, maintenance, professional, and technical support personnel are available in necessary numbers to ensure the safe and efficient operation of STPEGS. These Training Programs are described in site and/or Nuclear Training procedures, as appropriate. These programs shall be conducted by qualified station and/or vendor personnel.

HISTORICAL

13.2.1 Operator Training Program

13.2.1.1 Cold Licensed Operator Training. The training program for licensing Senior Reactor Operator (SRO) and Reactor Operator (RO) candidates provided a means of preparing these personnel for Nuclear Regulatory Commission (NRC) license examinations and subsequently, station operations.

Station operating and supervisory personnel who were required to qualify for license examinations were categorized by experience as follows:

1. Individuals with no previous nuclear experience
2. Individuals with nuclear experience at facilities not subject to licensing
3. Individuals holding, or who have held, licenses for comparable facilities

13.2.1.1.1 License Candidates with Fossil Power Plant Experience: License candidates with fossil power plant experience, but with no nuclear experience were given the following training:

1. Phase I – Nuclear Power Plant Fundamentals (11 weeks)

   This training was conducted at the Westinghouse Nuclear Training Center or through a combination of onsite classroom instruction and offsite research reactor training. Topics include:

   a. Nuclear Reactor Theory – 2 weeks
   b. Large PWR Core Physics – 2 weeks
   c. Health Physics, Instrumentation and Chemistry – 2 weeks
   d. Power Plant Systems and Engineering Concepts – 2 weeks
   e. Reactor Loading, Reactor Operations and Experiments – 3 weeks

HISTORICAL

2. Phase II – Operating PWR Training (10 weeks)

   This training consisted of a series of systems lectures. Phase II training was completed through a combination of onsite classroom instruction in plant specific systems and systematic observation training at an operating plant similar to STPEGS/or at the Westinghouse Nuclear Training Center with plant observation at the Zion Station of the Commonwealth Edison Company. Classroom lectures dealing with the Zion plant were followed by plant tours to observe the hardware and systems operations which have been discussed.

3. Phase III – Simulator Training (9 weeks)

   This training provided license candidates with actual plant operations and transient situations including startups, shutdowns, and operation under normal, off-normal, and emergency conditions. At the conclusion of simulator training, the candidates were given extensive RO and SRO written, oral, reactor startup, and simulator crew operating examinations. Upon successful completion of simulator training, the candidates received RO or SRO certification from Westinghouse. Phase III training was conducted on a simulator similar to STPEGS such as the Westinghouse Nuclear Training Center or on the STPEGS simulator.

HISTORICAL

4. Phase IV – Onsite Training

   Onsite training was conducted at STPEGS to include:

   a. Classroom Training (approximately 24 weeks)

      - Heat Transfer, Fluid Flow, and Thermodynamics – 2 weeks

      - Reactor Theory, Radiation Protection, and Chemistry – 4 weeks

      - Plant Systems and Procedures – 13 weeks – This series consisted of half-day classroom lectures with the remaining half day spent in the plant identifying equipment covered in the classroom presentation

      - Plant Accident and Transient Analysis – 1 week

      - Plant Administrative Procedures, Technical Specifications, and Conditions and Limitations of the Facility License – 1 week

      - Mitigating core Damage (MCD) – 1 week – As identified by H. Denton letter of March 28, 1980. This program (MCD) ensured that all operating personnel (Licensed Operators and appropriate managers, and technicians in the instrumentation and control [I&C] , health physics, and chemistry departments) received training commensurate with their responsibilities.

      Training program content included, but was not limited to , the following subjects with the above personnel receiving appropriate portions.

HISTORICAL

- Incore Instrumentation
- Excore Nuclear Instrumentation
- Vital Instrumentation
- Primary Chemistry
- Radiation Monitoring
- Gas Generation

b. Procedure Training (approximately 4 weeks)

This training was conducted onsite using actual or simulated control boards emphasizing:

HISTORICAL

- Normal Plant Operations and Integrated Plant Response – 2 weeks
- Abnormal and Emergency Plant Procedures – 2 weeks. This training was conducted on the STPEGS simulator.

HISTORICAL

c. On-The-Job Training (approximately 12 months)

In addition to classroom training, operating personnel received approximately 12 months of on-the-job training prior to plant startup. During this period, the Shift Managers and Unit Supervisors prepared written operating procedures and checkoff lists, prepared systems and equipment for pre-operating testing, and directed and planned the work assignments of the operating personnel. These personnel learned to operate the Nuclear Steam Supply System's support systems and simulated, as closely as possible, startup and shutdown of the reactor. Operating personnel were also actively engaged in pre-operational tests of actual equipment and systems.

5. Phase V – Pre-License Review and Audit

A 3 to 4 week pre-license review addressed the topics covered in Phase IV. In addition, an examination was administered in the same manner as the NRC examination as a final step in preparing license candidates for NRC Examination.

13.2.1.1.2 <u>License Candidates with Nuclear Power Plant Experience</u>: License candidates with Navy nuclear power plant experience (qualified as RO, EWS, ERS, PPWS, CRW, or EOOW positions listed in NUREG-1021, Section 109), but without commercial nuclear power plant experience, were given, as a minimum, a modified Phase III simulator training course, and Phases IV and V training were as described in Section 13.2.1.1.1, Items 4 and 5.

1. Phase I – Nuclear Power Plant Fundamentals

Military experience and academic training satisfied this requirement. The evaluation of license candidates during Phases III, IV, and V ensured that this assumption was valid.

2. Phase II – Operating PWR Training

   Military experience and training satisfied this requirement. The evaluation of license candidates during Phases III, IV, and V ensured that this assumption was valid.

3. Phase III – Simulator Training

   Military experience and training satisfied most of this requirement. However, candidates in this category received 2 weeks of simulator training in normal and off-normal commercial power plant operations.

HISTORICAL

4. Phase IV – Onsite Training

   Candidates received training as described in Section 13.2.1.1.1, Item 4.

5. Phase V – Pre-License Audit Review

   Candidates received training as described in Section 13.2.1.1.1, Item 5.

HISTORICAL

13.2.1.1.3 License Candidates Who Have Been Previously Licensed: License candidates who were previously NRC licensed as RO or SRO at a comparable reactor facility received, as a minimum, Phase IV and Phase V training as described in Section 13.2.1.1.1, Items 4 and 5. This included two weeks of simulator training in normal and off-normal power plant operations.

13.2.1.1.4 SRO Training: Training described in Section 13.2.1.1.1, Items 4 and 5 was conducted at the SRO level.

13.2.1.1.5 Instructor Qualifications: Instructors who taught safety systems, integrated plant response, transient, and simulator training successfully passed a NRC Senior Operator Instructor Certification Examination or equivalent. The qualifications of these instructors were maintained by their participation in the conduct of the Phase IV training program. Since a requalification program was not started until after Cold License, this participation was considered as enrollment in appropriate requalification program, as required by the NRC Harold Denton Letter (March 1980). Instructors participated in the requalification program once it was initiated.

13.2.1.1.6 Evaluation of Program Effectiveness: Progress of license candidates during the program was evaluated by written or oral examination to ensure adequate progress.

13.2.1.2 Replacement Training for NRC Licensed Plant Personnel and Non-Licensed Plant Operators.

13.2.1.2.1 Licensed Operators: The licensed operator replacement training program is based on a systematic approach to training. The licensed operator replacement and SRO upgrade program were initially accredited by INPO in June 1990 and meet the requirements of 10CFR55.

13.2.1.2.2 Non-Licensed Operators: The non-licensed operator replacement training program is based on a systematic approach to training. The non-licensed operator replacement program was initially accredited by INPO in June, 1990 and meets the requirements of 10CFR50.120.

13.2.1.2.3 Shift Manager: The Shift Manager training program is based on the INPO guidelines developed from industry task analysis for Shift Manager, and meets the requirements of 10CFR50.120.

13.2.1.3 Requalification Training for Licensed and Non-Licensed Operating Personnel.

13.2.1.3.1 Licensed Operators: The licensed operator requalification program is based on a systematic approach to training. The licensed operator and senior reactor operator upgrade program were initially accredited by INPO in June, 1990 and meet the requirements of 10CFR55.

13.2.1.3.2 Non-Licensed Operator Retraining Program: The non-licensed operator retraining program is based on a systematic approach to training. The non-licensed operator replacement training program was initially accredited by INPO in June, 1990 and meets the requirements of 10CFR50.120.

13.2.1.4 Shift Technical Advisor Training: The Shift Technical Advisor (STA) training program is based on a systematic approach to training. The STA program was initially accredited by INPO in June, 1990 and meets the requirements of 10CFR50.120.

13.2.2 Staff Training Programs

13.2.2.1 Maintenance Training.

13.2.2.1.1 I&C Training: The I&C maintenance training program is based on a systematic approach to training. This program was initially accredited by INPO in July 1990 and meets the requirements of 10CFR50.120.

13.2.2.1.2 Mechanical Maintenance Training: The mechanical maintenance training program is based on a systematic approach to training. This program was initially accredited by INPO in July 1990 and meets the requirements of 10CFR50.120.

13.2.2.1.3 Electrical Maintenance Training: The electrical maintenance training program is based on a systematic approach to training. This program was initially accredited by INPO in July 1990 and meets the requirements of 10CFR50.120.

13.2.2.2 Chemistry Analysis and Radiation Protection Training

13.2.2.2.1 Chemistry Analysis Training: The chemistry analysis training program is based on a systematic approach to training. This program was initially accredited by INPO in July 1990 and meets the requirements of 10CFR50.120.

13.2.2.2.2 Radiation Protection Training: The radiation protection training program is based on a systematic approach to training. This program was initially accredited by INPO in July, 1990 and meets the requirements of 10CFR50.120.

13.2.2.3 Engineering Support Personnel Training: The engineering support personnel training program is based on INPO guidelines developed from Industry wide task analysis that use a systematic approach to training. This program was initially accredited as the Manager and Technical Staff Training Program in June, 1990 and meets the requirements of 10CFR50.120.

13.2.2.4 Training for Mitigating Core Damage: Training is provided on the use of installed equipment and systems to control or mitigate accidents in which the core may be severely damaged. This program (MCD) ensures that operations personnel (Management, Licensed Operators, and STA) and appropriate managers and technicians in the Instrumentation and Control (I&C), Health Physics, and Chemistry Departments are trained commensurate with their responsibilities.

Training program content includes, but is not limited to, the following subjects with the personnel indicated above receiving appropriate portions:

Incore Instrumentation

Excore Instrumentation

Vital Instrumentation

Primary Chemistry

Radiation Monitoring

Gas Generation

13.2.2.5 General Employee Training. All persons permanently employed at STPEGS and who require unescorted access to the protected area are trained in the following areas commensurate with their job duties:

1. General Description of STPEGS

2. General Procedures and/or Instructions

3. Security Program

4. Industrial Safety Program

5. Fire Protection Program

6. Emergency Plan

7. Quality Assurance Program

8. Radiation Protection

Certain training requirements may be exempted in accordance with applicable station procedures. Temporary maintenance and service personnel shall be trained in the areas listed above to the extent necessary to assure safe execution of their duties.

13.2.2.6 Fire Protection Training. Fire protection training is conducted in accordance with the guidelines of 10CFR50, Appendix R and Branch Technical Position, Auxiliary and Power Conversion Systems Branch (APCSB) 9.5-1. This training includes classroom instruction, hands-on fire extinguishing, and plant drills. This program includes, but is not limited to, the following.

13.2.2.6.1 Fire Brigade Team Members:

1. Initial Training

   Initial instruction in the topics listed below is provided to each individual prior to assignment as a fire brigade member. The instruction includes such areas as:

   a. Idenfiication of the location and types of fire hazards that could produce fires within the plant, including identification of the toxic and corrosive characteristics of the products of combustion

   b. Identification of the location of installed and portable fire fighting equipment in each area and familiarization with the layout of the plant, including access and egress routes (initial training may be general in nature with more detail presented in retraining sessions)

   c. Proper use of available equipment and the correct method of fighting the following types of fire: electrical, cable and cable trays, hydrogen, flammable liquids, waste/debris, and file records

   d. Proper use of protective equipment including breathing, communication, lighting, and portable ventilation equipment

   e. General review of the format and use of preplans with particular emphasis on what equipment should be used in particular areas (detailed reviews shall be conducted during retraining sessions)

   f. Delineation of responsibilities and duties when working with offsite fire departments

   g. The proper method of fighting fires inside buildings and tunnels

2. Retraining

Planned meetings shall be held on a regularly scheduled basis of not less than four sessions a year for all brigade members to review changes in the fire protection program and other subjects as necessary.

Periodic refresher training sessions shall be held to repeat the classroom instruction program for all brigade members over a two-year period.

3. Practice Sessions

   Practice sessions shall be held for fire brigade members to familiarize them with the proper methods of fighting various types of fires, with practice in extinguishing actual fires, and in the use of protective equipment including emergency breathing apparatus. These sessions shall be conducted so as not to endanger safety-related equipment and shall be conducted at least once per year.

4. Drills

   At three-year intervals, a randomly selected, unannounced drill shall be critiqued by qualified individuals independent of the STPEGS staff. A copy of the report from such individuals shall be available for NRC review.

   Drills will address:

   a. The simulated use of equipment for the various situations and types of fires which could reasonably occur in safety-related areas

   b. Conformance, where possible, to the established fire preplans

   c. Operation of fire fighting equipment, where practicable including self-contained breathing apparatus, communication equipment, and portable and installed ventilation equipment

Drills shall be performed at least once per quarter for each shift fire brigade. At least one drill per year for each shift fire brigade shall be unannounced. Unannounced drills shall be scheduled a minimum of four weeks apart. At least one drill per year for each shift fire brigade shall be conducted on the backshift. Drills should be planned and conducted to established training objectives and critiqued to determine how well the training objectives were met. This critique should, as a minimum, assess: fire alarm effectiveness; response time; selection, placements, and use of equipment; the fire brigade leader's direction of the fire fighting effort; and each fire brigade member's response to the emergency.

Off-site fire departments shall be invited to participate in a drill, at least annually. Additionally or simultaneously, an annual plant evacuation drill shall be conducted.

13.2.2.6.2 Fire Brigade Team Leaders: The fire brigade team leaders and two brigade team members shall receive sufficient training in or have knowledge of plant safety-related systems to understand the potential effects of fire and fire suppressants on safe shutdown capability. The fire brigade team leaders shall also receive special instruction in directing and coordinating fire-fighting activities.

13.2.2.6.3 Fire Protection Personnel: Appropriate fire protection personnel shall be introduced to a program of specialized training. Instruction includes the following topics:

1. Analysis of building layout and system design with respect to fire protection requirements, including consideration of potential hazards associated with postulated design basis fires

2. Design and maintenance of fire detection, suppression, and extinguishing systems

3. Fire protection techniques and procedures

4. Training and manual fire fighting techniques and procedures for plant personnel and the fire brigade

13.2.2.6.4 Other Station Employees: At least once per year site employees shall be provided instruction on procedures for reporting fires. Security personnel shall be instructed in entry procedures for offsite fire departments, crowd control, and reporting potential fire hazards observed when touring the facility. Shift personnel shall be instructed in the actions expected of them in assisting the fire brigade in the event of a fire. Any or all of this training may be accomplished as part of the General Employee and/or Emergency Plan Retraining Programs.

Contract personnel, temporary employees, and visitors who are authorized unescorted access to the plant shall be given instruction to familiarize them with the plant evacuation signals, evacuation routes, and procedures for reporting fires.

13.2.2.6.5 Offsite Fire Departments: Training shall be made available to local fire departments regarding operational precautions when fighting fires at STPEGS. This training shall include an awareness of the need for radiological protection of personnel and the special hazards associated with STPEGS.

13.2.2.6.6 Construction Personnel: Training for construction personnel should include instruction in reporting fires, responding to alarms, and locating evacuation routes.

13.2.3 Applicable NRC Documents

The applicable portions of the NRC regulations, Regulatory Guides (RGs) and reports listed below will be used in providing guidance in the training of plant personnel.

13.2.3.1	10CFR50	"Licensing of Production and Utilization Facilities".
13.2.3.2	10CFR55	"Operators' License".
13.2.3.3	10CFR19	"Notices, Instructions and Reports to Workers; Inspections".
13.2.3.4	RG 1.8	"Personnel Selection and Training", see Table 3.12-1 for applicable revision number.
13.2.3.5	NUREG-0654	"Criteria for Preparation and Evaluation of Radiological Emergency Response Plan and Preparedness in support of Nuclear Power Plants".
13.2.3.6	RG 1.120	"Fire Protection Guidelines for Nuclear Power Plants", see Table 3.12-1 for applicability.
13.2.3.7	RG 8.2	"Guide for Administrative Practices in Radiation Monitoring", Revision 0.
13.2.3.8	RG 8.8	"Information Relevant to Maintaining Occupational Radiation Exposure As Low As Is Reasonably Achievable (Nuclear Power Reactors)", Revision 3.
13.2.3.9	RG 8.10	"Operating Philosophy for Maintaining Occupational Exposures As Low As Is Reasonably Achievable", Revision 1-R.
13.2.3.10	NUREG-0731	"Guidelines for Utility Management Structure and Technical Resources" (Draft report for interim use and comment, September 1980).
13.2.3.11	RG 8.29	"Instruction Concerning Risks from Occupational Radiation Exposure", Revision 1.
13.2.3.12	NUREG-0737	"Clarification of TMI Action Plan Statements".
13.2.3.13	ANS/ANSI 3.1-1981	"Standard for Qualification and Training of Personnel for Nuclear Power Plants"-.
13.2.3.14	NRC Harold Denton Letter	(March 1980) to All Power Reactor Applicants and Licenses, "Qualifications of Reactor Operators".
13.2.3.15	ANSI/ANS-18.1-1971	"Standard for Qualification and Training of Personnel for Nuclear Power Plants".
13.2.3.16	ANSI/ANS-3.5-2009	"Nuclear Power Plant Simulators for Use in Operator Training and Examination".

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13.3 EMERGENCY PLANNING

A comprehensive emergency plan for the South Texas Project Electric Generating Station, Units 1 and 2, is provided as a separate volume to this application.

13.4 REVIEW AND AUDIT

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13.4.1 Onsite Review

Onsite review activities are performed by the Plant Operations Review Committee (PORC), which is described in Chapter 19.0 of the Operations Quality Assurance Plan (OAQP).

13.4.2 Independent Review

Independent Review requirements are described in Chapter 19.0 of the Operations Quality Assurance Plan (OAQP).

13.4.3 Audit Program

The Senior Management Team (SMT) is cognizant of audits of activities described in the OQAP. The procedures and organization employed to implement the audit program are discussed in Chapters 15.0 and 19.0 of the OQAP. The Quality staff conducts independent overview activities as defined in the OQAP. Audits of safety-related activities are planned in accordance with a written plan and performed with a frequency commensurate with their safety significance, past performance, and regulatory requirements. Qualification of auditors, conduct of the audit, reporting of the audit, and follow-up of audit deficiency corrections are controlled by a written procedure. Copies of audit reports are distributed to the SMT.

13.5 PLANT PROCEDURES

Administrative and operating procedures will be utilized by the plant staff in the performance of their duties to ensure that routine operating, off-normal and emergency activities are conducted in a manner that protects the health and safety of the general public and plant personnel, and maintains the reliability and integrity of plant equipment.

13.5.1 Administrative Procedures

13.5.1.1 Conformance With Regulatory Guide 1.33. Regulatory Guide (RG) 1.33, Rev. 2, will be used as a guide for the preparation of plant administrative policies and procedures except the requirement in ANSI N18.7-1976 to perform reviews no less frequently than every two years. The requirements of this guide will be met for those systems and components indicated in Section 3.2, to which 10CFR50 Appendix B requirements are applied.

13.5.1.2 Preparation of Procedures. Safety-related operations will be conducted by detailed approved procedures. Safety-related Administrative procedures will be reviewed by the Plant Operations Review Committee (PORC). After the Plant Manager's final review and approval, procedures will be issued for use. Procedure preparation and review is an ongoing effort throughout the life of the plant. Operating procedures required to load fuel were completed approximately six months prior to fuel load.

13.5.1.3 Procedures.

1. Conduct of Operations: The responsibilities and authorities of station operating personnel are determined through administrative procedures in the form of conduct-of-operations procedures. As a minimum, these procedures will establish the following:

   a. Responsibility and authority of the Reactor Operator (RO) to manipulate controls which directly affect core reactivity and/or the manipulation of apparatus and mechanisms other than controls which may affect the reactivity or power level of a reactor, including a reactor trip if he deems it necessary. He is also assigned the responsibility for knowing what safety-related equipment and systems have limits and setpoints specified in the plant Technical Specifications or Operating Procedures.

   b. Responsibility and authority of the Shift Manager for licensed activities at the unit under his control. There may be additional operating personnel on shift holding senior reactor operator licenses, but the Shift Manager is delineated the authority of the senior reactor operator (SRO), pursuant to 10CFR50.54(1). In the event of the Shift Manager's absence, this authority will be designated to another SRO.

   c. Requirements for the presence of a licensed RO or SRO "at the controls" at all times during the operation of a unit, pursuant to 10CFR50.54(k). Requirements for the presence of a licensed SRO at the facility pursuant to 10CFR50.54(m). The area of the control room designated as "at the controls" will be specified.

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d. Requirements that no one is permitted to manipulate facility controls which affect reactivity who is not a licensed RO or SRO, except for license trainees operating under the direction of a licensed operator, pursuant to 10CFR50.54 (i). Procedures require that all station personnel operating plant apparatus and mechanisms (other than controls) which may affect the reactivity or power level of a unit must notify and obtain the consent of the reactor operator prior to initiating such action, pursuant to 10CFR50.54 (j).

e. Responsibilities and authorities of the operating shift, pursuant to NUREG-578, Item 2.2.1a.

f. Procedures for shift relief and turnover pursuant to NUREG-0694, TMI-Related Requirements for New Operating Licenses, item I.C.2., with the exception that Maintenance personnel do not use signed checklists and logs.

g. Limits on working hours pursuant to Title 10, Part 26, "Fitness For Duty Programs," of the Code of Federal Regulations, Subpart I, "Managing Fatigue."

h. Procedures for control room access pursuant to NUREG-0694, item I.C.4.

i. Procedure for the feedback of operating information pursuant to NUREG-0737, item I.C.5.

j. Procedures for verifying the correct performance of operating activities pursuant to NUREG-0737, item I.C.6.

k. Controls governing crane operations including requirements that crane operators who operate cranes over fuel pools be qualified and conduct themselves pursuant to the guidelines of ANSI B30.2-1976, Chapter 2-3.

2. Equipment Control: Authorization for release of safety-related equipment or systems for maintenance will be granted by designated operating personnel holding a SRO license. Such authorization will be documented and will be based upon verification that the equipment or system may be taken out of service, how long it may remain out of service, and to what degree redundant safety systems may be degraded by removing the equipment. After permission has been granted to remove equipment from service, it will be isolated to provide protection for plant personnel and equipment. When entry into a closed system is required, control measures will be established to prevent entry of extraneous material and to assure that foreign material is removed before the system is reclosed. Equipment tagging will be utilized to secure and identify equipment in a controlled status. Temporary modifications, such as temporary bypass lines, electrical jumpers, lifted electrical leads, and temporary trip settings will be performed and documented in accordance with approved procedures. When equipment is ready to be returned to service, operating personnel will place the equipment in operation and verify and document its functional acceptability. Safety-related equipment will have proper alignment independently verified by a second qualified person or functional testing unless such verification would result in significant radiation exposure.

3. Maintenance Control: Procedures that govern the work process program will be reviewed by the PORC to assure adequate guidelines are in place which ensure that the equipment is returned to a state of quality at least equivalent to that specified originally. Maintenance or modification of plant equipment which may affect the functioning of safety-related structures, systems, or components will be performed under approved work packages. Work packages will be developed in accordance with the approved work process program.

   Maintenance or modification of plant equipment which does not affect the functioning of safety-related structures, systems, or components may be performed in accordance with approved procedures, documented instructions, approved drawings, or appropriate sections of related vendor materials which provide adequate guidance to assure the required quality of work.

   Those skills normally possessed by qualified maintenance personnel will not require detailed step-by-step written instructions.

4. Surveillance Schedule: Surveillance testing will be prescribed by administrative procedures to ensure that the reliability of plant safety-related structures, systems, and components is maintained in accordance with station Technical Specifications. Department Managers are responsible for ensuring that surveillances assigned to their departments are performed in accordance with the master surveillance schedule.

5. Logbook Control: Administrative procedures will prescribe the usage, control, and number of logbooks. These procedures will establish provisions for the preparation and retention of the logbooks including responsibility for maintaining logbooks and storing them at specified locations.

6. Temporary Procedures: Temporary administrative procedures may be issued during the operational phase of the station for the performance of activities which are of a nonrecurring nature. Such activities may include such items as:

   a. Direction of operations during testing, refueling, maintenance, and modifications

   b. Guidance in unusual situations not within the scope of the normal procedures CN-3101

   c. Ensuring orderly and uniform operations for short periods when the plant, a system, or a component of a system is performing in a manner not covered by existing procedures

   Limitations involved in the use of the temporary procedure, such as the time interval during which it is in force, will be clearly stated on the procedure. Safety-related temporary administrative procedures will be reviewed by the PORC.

7. Fire Control Procedures: Administrative procedures will be used for preventing, detecting, suppressing, and extinguishing plant fires. These administrative procedures will describe:

   a. The method for reporting a fire

b. The method for obtaining permits for cutting, welding, and open flame work

c. The method for controlling the movement of combustible material within the plant.

d. The relationships of the fire fighting command with the plant staff, outside fire departments, and construction forces

e. The establishment of a fire watch

f. The organization and duties of the fire brigade

13.5.2 Operating and Maintenance Procedures

13.5.2.1 Control Room Operating Procedures. Operating activities which may affect the proper functioning of the station's safety-related systems, or components will be performed in accordance with approved procedures. These procedures will provide a preplanned method of conducting operations of systems in order to eliminate errors due to on-the-spot analyses and judgments. Procedures will be sufficiently detailed so that qualified individuals, can perform the required functions without direct supervision. Written procedures cannot cover all possible contingencies and therefore must contain a certain degree of flexibility. The plant operating procedures that will be prepared initially are identified below.

1. System Operating Procedures

System operating procedures will provide instructions for alignment, energizing, filling, venting, draining, starting up, shutting down, and/or changing modes of operation as well as other operating instructions. Safety-related operating procedures will be prepared for the systems and equipment listed below:

a. Reactor Coolant System

b. Rod Control System

c. Residual Heat Removal System

d. Emergency Core Cooling System

e. Containment Spray System

f. Containment Heating and Ventilation System

g. Containment Purge System

h. Spent Fuel Pool Cooling and Cleanup System

i. Main Steam System

j. Feedwater System (feedwater pumps to steam generator)

k. Auxiliary Feedwater System

l. Essential Cooling Water System

m. Chemical and Volume Control System

n. Mechanical Auxiliary Building Heating and Ventilation System

o. Control Room and Electrical Auxiliary Building Heating and Ventilation Systems

p. Fuel Handling Building Heating and Ventilation System

q. Instrument Air System

r. Electrical Systems

   1) Offsite (circuits between the offsite transmission network and the onsite Class 1E distribution system)

   2) Onsite

      a) Class 1E Emergency Power Sources

      b) Class 1E AC Distribution System

      c) Class 1E DC Distribution System

s. Nuclear Instrumentation System

t. Solid State Protection System

u. Hydrogen Recombiner

v. Component Cooling Water System

w. Essential Chilled Water System

x. RCB Chilled Water System

y. Emergency Diesel Generator

2. General Operating Procedure

General operating procedures will provide instructions for operating the station as a whole during major evolutions or steady state conditions. The following is the minimum list of plant general operating procedures.

a. Plant Heat-up

b. Secondary Plant Startup

c. Reactor Startup

d. Plant Startup to 100 Percent

e. Plant Shutdown from 100 Percent to Hot Shutdown

f. Plant Cooldown

3. Off-Normal Operating Procedures

Off-normal operating procedures will provide the necessary instructions for restoring an operating variable to its normal controlled value when it departs from its range or to restore normal operating conditions following a perturbation which could potentially degrade into an emergency or violate plant technical specifications if proper action were not taken. Each procedure will identify the symptoms of the off-normal condition, automatic actions that may occur, and the appropriate immediate and subsequent operator actions to be taken. The following is the minimum list of plant off-normal operating procedures.

a. Loss of Instrument Air

b. Loss of any 13.8 kV or 4.16 kV Bus

c. Loss of Component Cooling Water System

d. Loss of Charging

e. Loss of Condenser Vacuum

f. Malfunction of the Reactor Coolant Makeup System

g. Control Rod Malfunction

h. Loss of Main Feedwater

i. Control Room Evacuation

j. Loss of Residual Heat Removal

k. High Reactor Coolant Activity

l. Loss of Automatic Pressurizer Control

m. Turbine Load Rejection

n. Dropped or Damaged Fuel Assembly

4. Emergency Operating Procedures

Emergency operating procedures will provide the necessary instructions to guide operations during conditions which might lead to injury of plant personnel or the general public and which might possibly lead to the release of radioactivity in excess of established limits. These procedures will be symptom oriented and designed to maintain the critical safety functions of the plant. STPEGS is a participant in the Westinghouse Owners Group (WOG) and will use the Emergency Response Guidelines developed by the WOG in response to NUREG-0737, Item I.C.1 as a basis for the STPEGS Emergency Operating Procedures. Thus, the STPEGS Emergency Operating Procedures will comply with the intent of NUREG-0737, Item I.C.1 and NUREG-0660, Items I.C.8 and I.C.9. In addition, the STPEGS Emergency Operating Procedures will meet the intent of NUREG-0799.

5. Annunciator Response Procedures

Annunciator response procedures will specify operator actions necessary to respond to an abnormal condition as indicated by an alarm. These procedures will include alarm setpoints, probable causes, automatic actions, immediate manual actions, subsequent actions, and applicable references. In order to ensure that control room annunciator response procedures are readily accessible for reference, an annunciator response guide will be employed to permit easy retrieval. The procedures will be grouped within the guide by annunciator panel number. The procedures will be further subdivided by row and column number upon each panel so that the response procedure for any annunciator may be quickly located. The annunciator response guide will be maintained in the control room. The Plant Operations Manager will be responsible for maintaining the guide up-to-date. Since the number of these annunciator response procedures is so large, they will not be listed in the Updated Final Safety Analysis Report (UFSAR).

6. Temporary Operating Procedures

Temporary operating procedures will provide instructions for plant operations which are of a nonrecurring nature such as:

a. The direction of activities during special testing or maintenance

b. Guidance in unusual situations not within the scope of normal procedures

c. Assuring orderly and uniform operations for short periods of time when the station, a system, or component is performing in a manner not covered by existing procedures

d. When modifications are made such that portions of the existing procedures do not apply

13.5.2.2 Other Procedures. Maintenance, implementation of Graded Quality Assurance and Comprehensive Risk Management, and other activities which may affect the proper functioning of the station's safety-related structures, systems, or components will be performed in accordance with approved procedures. These procedures will provide a preplanned method of conducting activities in order to eliminate errors. They will be sufficiently detailed so that qualified individuals can perform the required functions without direct supervision. However, approved procedures cannot cover all contingencies and therefore must contain a certain degree of flexibility. The general character and objectives of these procedures are described below.

1. Radiation Protection Procedures

   Radiation protection procedures will, along with design shielding, limit the exposure of plant personnel to airborne radioactivity and radiation during plant operation and maintenance. They will be incorporated into the Plant Procedures Manual by the Health Physics Manager and his staff prior to initial plant startup. These procedures will include personnel access control , air-monitoring program, routine radiological surveys, and the utilization of portable instrumentation. Use of these procedures, along with a careful monitoring program, a personnel training program, proper work procedures, and the use of special equipment, will ensure that plant personnel receive less than the radiological exposure limits presented in 10CFR20.

2. Emergency Preparedness Procedures

   Information concerning these procedures is presented in the South Texas Project Electric Generating Station Emergency Plan.

3. Instrument and Control Procedures

   Instrument and Control procedures will provide detailed instructions for the proper calibration, testing, and adjustment of all safety-related instrumentation and control systems. They will ensure measurement accuracies adequate to maintain plant safety-related parameters within safety limits. The Nuclear Group will be responsible for developing these procedures. The plant Instrumentation and Control (I&C) Division under the supervision of the I&C Manager, will be responsible for implementing these procedures.

4. Chemistry Procedures

   Chemistry procedures will provide instructions to control chemistry-and radiochemistry-related activities. They will be developed and implemented by the Chemistry Division under supervision of the Manager, Chemistry. These procedures will be incorporated into the Plant Procedures Manual and will include such instructions as:

   a. The nature and frequency of sampling and analysis to be performed

   b. Prescribed limits for coolant quality

c. Limitations on concentrations of corrosive agents which could become sources of radiation hazards

d. Treatment and control of radioactive wastes and control of radioactive calibration sources

5. Radioactive Waste Management Procedures

Procedures for the operation of the Radwaste Processing Systems will, in conjunction with the plant Radiation Protection Procedures, provide for the control, treatment, and management of radioactive wastes on the plant site. These procedures will be developed and implemented by the Plant Chemical Operations Section under supervision of the Chemical Operations Supervisor.

6. Electrical and Mechanical Maintenance Procedures

Electrical and Mechanical Maintenance Procedures will provide detailed instructions, where applicable, to ensure that electrical and mechanical work is performed safely, correctly, and in accordance with prescribed radiation protection measures. The Maintenance Department will be responsible for developing these procedures. These procedures will be implemented by the Electrical and Mechanical Maintenance Divisions under the supervision of the Electrical Maintenance Manager and Mechanical Maintenance Manager respectively.

7. Material Control Procedures

Information concerning these procedures is presented in the Operations Quality Assurance Plan.

8. Nuclear Security Procedures

Information concerning these procedures is presented in the South Texas Project Electric Generating Station Security Procedures.

9. Fire Protection Procedures

Fire protection procedures will be written to ensure a coordinated effort in preventing, detecting, suppressing, and extinguishing plant fires. Included will be:

a. Housekeeping practices

b. General instructions for plant personnel in the event of a fire

c. Procedures for maintenance, testing, and calibration of the fire detection system

d. Procedures for using fire water systems, portable fire fighting equipment and other installed fire suppression systems

e. Classification and methods of combatting fires

f. Instructions for fighting fires in various plant areas.

10. Prerequisite and Preoperational Testing Procedures

A detailed description of the Prerequisite and Preoperational Testing Program, including personnel qualification, is in Section 14.2.

11. Initial Startup Test Procedures

Initial Startup Test Procedures will be written to provide detailed instructions for eh conduct and coordination of the Initial Startup Test Program. Initial Startup Test personnel who are performing, reviewing, and evaluating activities related to the Initial Startup Test Program are qualified to perform such activities through previous experience and/or training. The qualifications of these personnel conform to the requirements specified in RG 1.8.

12. Measurement and Test Equipment Procedures

Measurement and Test Equipment (M&TE) procedures will provide instructions for the proper calibration, maintenance, testing, and adjustment of measurement and test equipment used in activities affecting the quality of safety-related systems. The procedures will conform to the requirements of the Operations Quality Assurance Program (OQAP). The Metrology Division under the supervision of the Metrology Laboratory Manager, will be responsible for developing and implementing these procedures.

13. Graded Quality Assurance and Comprehensive Risk Management Procedures

Comprehensive Risk Management is a process by which the risk to station personnel and the public's health and safety are evaluated as a result of changes in commitments, processes, activities, human and equipment performance. Graded Quality Assurance is the process by which risk-informed methodology and performance-based information and analyses are combined to establish appropriate level of programmatic controls for systems, components or activities and appropriate levels of independent oversight needed to provide necessary assurance that items will operate safely and activities are accomplished as prescribed. Changes to the implementing procedures for Grade Quality Assurance and the Comprehensive Risk Management Program are reviewed in accordance with the requirements of 10CFR50.59.

The Configuration Risk Management Program, which is part of the Comprehensive Risk Management Program, is used to assess the risk impact of equipment out of service and to maintain station risk at the desired levels. The Configuration Risk Management Program is used to assess risk impacts for planned and unplanned equipment outages and is the primary tool for performing the risk assessment required by 10CFR50.65(a)(4).

13.6 NUCLEAR SECURITY

Security plans for South Texas Project Electric Generating Station, Units 1 and 2, are provided separate from this report. These security plans consist of the composite security plan (physical security, training and qualification, and safeguards contingency plans) and the cyber security plan. CN-3244

13.7 RISK-INFORMED SPECIAL TREATMENT REQUIREMENTS

13.7.1 Introduction

NRC regulations in 10 CFR Parts 21, 50, and 100 contain special treatment requirements that impose controls to ensure the quality of components that are safety-related, important to safety, or otherwise come within the scope of the regulations. These special treatment requirements go beyond normal commercial and industrial practices, and include quality assurance (QA) requirements, qualification requirements, inspection and testing requirements, and Maintenance Rule requirements. STP has been granted an exemption from the special treatment requirements. Table 13.7-1 identifies the regulations from which an exemption was granted and the scope of the exemption. This exemption only pertains to special treatment requirements; it does not change the requirements of 10 CFR Parts 50 and 100 that specify design or functional requirements for SSCs; i.e., the requirements that specify the safety functions to be performed by a system or component (including design features to prevent adverse impacts upon the safety function of one SSC due to the failure of another SSC). Also it does not change any design or functional requirements in the other sections of the STP UFSAR or requirements of the STP Technical Specifications.

STP has a risk-informed process for categorizing the safety/risk significance of components. This process is described in Section 13.7.2. Components with no or low safety significance have been exempted from the scope of most of the NRC regulations that impose special treatment requirements, and instead are subject to normal industrial and commercial practices. Additionally, non-safety-related components (and, under certain circumstances, safety-related components) with medium or high safety significance are evaluated for enhanced treatment. Components retain their original regulatory requirements unless they have been categorized using the process described below. The treatment for the various categories of components is described in Section 13.7.3. As part of this process, STP also performs continuing evaluations and assessments, which are described in Section 13.7.4. Finally, STP applies quality assurance to this process, and controls changes to the process, as described in Section 13.7.5.

13.7.2 Component Categorization Process

13.7.2.1 Overview of Categorization Process. The process utilized by STP in categorizing components consists of the following major tasks:

1. Identification of functions performed by the subject plant system.
2. Determination of the risk significance of each system function.
3. Identification of the system function(s) supported by that component.
4. Identification of a risk categorization of the component based on probabilistic risk assessment (PRA) insights (where the component is modeled)
5. Development of a risk categorization of the component based on deterministic insights.

6. Designation of the overall categorization of the component, based upon the higher of the PRA categorization and the deterministic categorization.
7. Identification of critical attributes for components determined to be safety/risk significant.

The processes for determining the PRA risk categorization and the deterministic risk categorization of a component are described in more detail in Sections 13.7.2.3 and 13.7.2.4. Additionally, the process for categorizing the pressure boundary function of ASME components is described in Section 13.7.2.5.

Based upon these processes, a component is placed into one of four categories: 1) high safety/risk significant (HSS), 2) medium safety/risk significant (MSS), 3) low safety/risk significant (LSS), and 4) non-risk significant (NRS). The terms HSS, MSS, and LSS are synonymous with the risk categorization terms of High, Medium, and Low, respectively. This categorization process does not, in and of itself, affect the other classifications of the component (e.g., safety, seismic, ASME classification).

The process is implemented by individuals experienced in various facets of nuclear plant operation. This integrated decision-making process is described in more detail in Section 13.7.2.2.

13.7.2.2 Comprehensive Risk Management Process. The integrated decision-making process used by STP is controlled by procedure. This process incorporates the use of experienced individuals who apply risk insights and judgement to categorize components in accordance with the process described in this Section.

The designated individuals have expertise in the areas of risk assessment, operations, maintenance, engineering, quality assurance, and licensing, including at least three individuals with a minimum of five years experience at STP or similar nuclear plants, and at least one individual who has worked on the modeling and updating of the PRA for STP or similar plants for a minimum of three years.

Management review of the integrated decision-making process is performed to ensure effective implementation of the process.

Procedures control the identification of and processes used by the designated individuals. Procedures also identify training requirements for the designated individuals, including training on probabilistic risk assessment, risk ranking, and the graded quality assurance process. In addition, the procedures specify the requirements for a quorum, meeting frequencies, the decision-making process for determining the categorization of components, the process for resolving differing opinions, and periodic reviews of the appropriateness of the programmatic control and oversight of categorized components. Finally, procedures control the management review activities.

13.7.2.3 PRA Risk Categorization Process. A component's risk categorization is initially based upon its impact on the results of the PRA. STP's PRA calculates both core damage frequency (CDF) and containment response to a core damaging event, including large early release frequency (LERF). The PRA models internal initiating events at full power, and also accounts for the risk associated with external events.

The PRA configuration control program incorporates a feedback process to update the PRA model. The updates are segregated into two categories:

- The plant operating update incorporates plant design changes and procedure changes that affect PRA modeled components, initiating event frequencies, and changes in SSC unavailability that affect the PRA model. These changes will be incorporated into the model on a period not to exceed 36 months.

- The comprehensive data update incorporates changes to plant-specific failure rate distributions and human reliability, and any other database distribution updates (examples would include equipment failure rates, recovery actions, and operator actions). This second category will be updated on a period not to exceed 60 months.

The PRA model may be updated on a more frequent basis.

Only components that are modeled in the PRA are given an initial risk categorization. The PRA risk categorization of a component is based upon its Fussell-Vessely (FV) importance, which is the fraction of the CDF and LERF to which failure of the component contributes, and its risk achievement worth (RAW), which is the factor by which the CDF and LERF would increase if it were assumed that the component is guaranteed to fail. Specifically, PRA risk categorization is based upon the following:

For individual component failures: CN-2959

PRA Ranking	**Criteria**
High	RAW ≥ 100.0 or FV ≥ 0.01 or FV ≥ 0.005 and RAW ≥ 2.0
Medium (Further Evaluation is Required)	FV < 0.005 and 100.0 > RAW ≥ 10.0
Medium	FV ≥ 0.005 and RAW < 2.0 or FV < 0.005 and 10.0 > RAW ≥ 2.0
Low	FV < 0.005 and RAW < 2.0

For common cause component failures:

PRA Ranking	Criteria
High	CCF RAW ≥ 100.0 or FV ≥ 0.01 or FV ≥ 0.005 and CCF RAW ≥ 20.0
Medium	FV ≥ 0.005 and CCF RAW <20.0 or FV < 0.005 and 100.0 > CCF RAW ≥20.0
Low	FV < 0.005 and CCF RAW < 20.0

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To determine the impact of a potential change in reliability of the LSS components on the overall plant risk, a sensitivity study is performed as part of the periodic updates to the PRA to determine the cumulative impact on CDF and LERF from postulating a factor of 10 increase in the failure rates for all modeled LSS components and non-categorized low ranking PRA components. The increases in CDF and LERF are determined to be acceptable using the guidelines for changes as outlined in Regulatory Guide 1.174.

To address defense-in-depth issues related to Late Containment Failures, a similar sensitivity analysis is performed as part of the periodic updates to the PRA. This study postulates an increase in component failure rates by a factor of 10 for all modeled LSS components and non-categorized low ranking PRA components. STP compares the resulting late containment failure frequency with its nominal frequency to assure that the delta increase in the late containment failure frequency is small, in support of adhering to the defense-in-depth philosophy stated in Regulatory Guide 1.174.

13.7.2.4 Deterministic Categorization Process. Components are subject to a deterministic categorization process, regardless of whether they are also subject to the PRA risk categorization process. This deterministic categorization process can result in an increase, but not a decrease (from the PRA risk), in a component's categorization.

A component's deterministic categorization is directly attributable to the importance of the system function supported by the component. In cases, where a component supports more than one system function, the component is initially classified based on the highest deterministic categorization of the function supported. In categorizing the functions of a system, five critical questions regarding the function are considered, each of which is given a different weight. These questions and their weight are as follows:

QUESTION	WEIGHT
Is the function used to mitigate accidents or transients?	5
Is the function specifically called out in the Emergency Operating Procedures (EOPs) or Emergency Response Procedures (ERPs)?	5

Does the loss of the function directly fail another risk-significant system?	4
Is the loss of the function safety significant for shutdown or mode changes?	3
Does the loss of the function, in and of itself, directly cause an initiating event?	3

Based on the impact on safety if the function is unavailable and the frequency of loss of the function, each of the five questions is given a numerical answer ranging from 0 to 5. This grading scale is as follows:

"0" - Negative response

"1" - Positive response having an insignificant impact and/or occurring very rarely

"2" - Positive response having a minor impact and/or occurring infrequently

"3" - Positive response having a low impact and/or occurring occasionally

"4" - Positive response having a medium impact and/or occurring regularly

"5" - Positive response having a high impact and/or occurring frequently

The definitions for the terms used in this grading scale are as follows:

Frequency Definitions –

- Occurring Frequently - continuously or always demanded
- Occurring Regularly - demanded > 5 times per year
- Occurring Occasionally - demanded 1-2 times per cycle
- Occurring Infrequently - demanded < once per cycle
- Occurring Very Rarely - demanded once per lifetime

Impact Definitions –

- High Impact - a system function is lost which likely could result in core damage and/or may have a negative impact on the health and safety of the public

- Medium Impact - a system function is lost which may, but is not likely to, result in core damage and/or is unlikely to have a negative impact on the health and safety of the public

- Low Impact - a system function is significantly degraded, but no core damage and/or negative impact on the health and safety of the public is expected

- Minor Impact - a system function has been moderately degraded, but does not result in core damage or negative impact on the health and safety of the public

- Insignificant Impact - a system function has been challenged, but does not result in core damage or negative impact on the health and safety of the public

Although some of these definitions are quantitative, both of these sets of definitions are applied based on collective judgment and experience.

The numerical values, after weighting, are summed; the maximum possible value is 100. Based on the sum, functions are categorized as follows:

SCORE RANGE	CATEGORY
0 - 20	NRS
21 - 40	LSS
41 - 70	MSS
71 - 100	HSS

A function with a low categorization due to a low sum can receive a higher deterministic categorization if any one of its five questions received a high numerical answer. Specifically, a weighted score of 25 on any one question results in an HSS categorization; a weighted score of 15-20 on any one question results in a minimum categorization of MSS; and a weighted score of 9-12 on any one question results in a minimum categorization of LSS. This is done to ensure that a function with a significant risk in one area does not have that risk contribution masked because of its low risk in other areas.

In general, a component is given the same categorization as the highest categorized system function that the component supports. However, a component may be ranked lower than the associated system function based upon diverse and/or multiple independent means available to satisfy the system function.

General notes may be used to document component risk justification for similar component types that are treated the same from system to system. Components covered by a general note are evaluated to ensure proper applicability of the note and appropriateness of the risk categorization. The use of general notes is an administrative tool that allows for increased efficiency in the documentation of justifications of large numbers of similar components. General notes are not used for categorizing system functions.

13.7.2.5 Categorization of the Pressure Boundary Function of ASME Components

In addition to the results of the categorization process discussed in Sections 13.7.2.3 and 13.7.2.4 above, STP considers other information in categorizing the pressure boundary function of ASME components. Specifically, for ASME Class 1 and 2 components, STP has established a risk ranking process in conjunction with its relief requests for risk-informed inservice inspection (RI-ISI) under NRC Regulatory Guide 1.178, “An Approach for Plant-Specific Risk-Informed Decisionmaking: Inservice Inspection of Piping.” This process is based on the NRC-endorsed EPRI RI-ISI methodology. For ASME Class 3 components, STP will follow this RI-ISI methodology for risk ranking. STP will apply this methodology to Class 3 systems or portions of systems for which the exemption from 10CFR 50.55a(g) is desired.

The RI-ISI methodology for risk ranking applies only to piping. STP assigns other components the same pressure boundary risk rank as the associated section of piping, or performs a technical evaluation that supports a lower pressure boundary risk rank based on such factors as differences in design features and/or degradation mechanisms that are less severe for these components than for the associated piping.

For determining the final pressure boundary category of ASME components for purposes of the exemption from 10 CFR 50.55a(g), STP uses the higher of the RI-ISI risk ranking or the categorization of the pressure boundary function determined by the process discussed in Section 13.7.2.4. Supports are assigned the same category as the final pressure boundary category of the highest ranked piping or component within the piping analytical model in which the support is included.

In order to provide additional assurance, STP performs periodic tests, up to and including tests equivalent to ASME Section XI tests, to ensure that the pressure boundary of LSS and NRS components is sufficiently maintained.

13.7.2.6 Defense-in-Depth and Safety Margins. For the following reasons, the exemption and the categorization process maintain defense in depth and sufficient safety margins:

- Design and functional requirements of systems will not be changed by this exemption.
- No existing plant barriers are removed or altered.
- Design provisions for redundancy, diversity, and independence are maintained.
- The plant’s response to transients or other initiators is not affected.
- Preventive or mitigative capability of components is preserved.
- There is no change in any of the safety analyses in the UFSAR.
- Existing safety-related LSS and NRS components will not be replaced, absent good cause (e.g., obsolescence or failure). Since the existing safety-related LSS and NRS components were designed, procured, manufactured, and installed in accordance with the existing special treatment requirements, these components have inherent design margins to perform their intended functions that will not be adversely affected by this exemption.
- The treatment processes described in Section 13.7.3 provide an appropriate and acceptable level of confidence that safety-related LSS and NRS components will be able to perform their intended functions.

- The corrective action program is applied to safety-related LSS and NRS components. This program provides reasonable confidence that deficiencies involving safety-related LSS and NRS components will be identified and corrected, and necessary action is taken to ensure acceptable performance levels are maintained.

13.7.3 Treatment for Component Categories

13.7.3.1 Description of Treatment for Component Categories. The following treatment is provided for the various component categories:

- Safety-Related HSS and MSS Components - The purpose of treatment applied to safety-related HSS and MSS SSCs is to maintain compliance with NRC regulations and the ability of these SSCs to perform risk-significant functions consistent with the categorization process. These components continue to receive the treatment required by NRC regulations and STP's associated implementing programs.

  Some safety-related components may be called upon to perform functions that are beyond the design basis or perform safety-related functions under conditions that are beyond the design basis. STP's PRA does not take credit for such functions unless there is a basis for confidence that the component will be able to perform the functions (e.g., demonstrated ability of the component to perform the functions under the specified conditions). If STP takes credit for such functions beyond that described above, STP would use the process described in Section 13.7.3.2 to evaluate these risk-significant functions that are not being treated under STP's current programs.
- Non-Safety-Related HSS and MSS Components - The purpose of treatment applied to non-safety-related HSS and MSS SSCs is to maintain their ability to perform risk-significant functions consistent with the categorization process. These components will continue to receive any existing special treatment required by NRC regulations and STP's associated implementing programs. Additionally, the risk-significant functions of these components will receive consideration for enhanced treatment. This consideration is described in Section 13.7.3.2.

- Safety-Related LSS and NRS Components - These components receive STP's normal commercial and industrial practices. These practices are described in Section 13.7.3.3.

- Non-Safety-Related LSS and NRS Components - The treatment of these components is not subject to regulatory control.

- Uncategorized Components - Until a component is categorized, it continues to receive the special treatment required by NRC regulations and STP's associated implementing programs, as applicable.

13.7.3.2 Enhanced Treatment for HSS and MSS Components. Non-safety-related HSS and MSS components may perform risk-significant functions that are not addressed by the special treatment requirements in NRC regulations or STP's current treatment programs.

When a non-safety-related component is categorized as HSS or MSS, STP documents the condition under the corrective action program and determines whether enhanced treatment is warranted to enhance the reliability and availability of the function. In particular, STP evaluates the treatment applied to the component to ensure that the existing controls are sufficient to maintain the reliability and availability of the component in a manner that is consistent with its categorization. This process evaluates the reliability of the component, the adequacy of the existing controls, and the need for any changes. If changes are needed, additional controls are applied to the component. In addition, the component is placed under the Maintenance Rule monitoring program, if not already scoped in the program (i.e., failures of the component are evaluated and Maintenance Rule Functional Failures (MRFF) involving the component are counted against the performance criteria at the plant/system/train level, as applicable). Additionally, as provided in the approved Graded Quality Assurance (GQA) program, non-safety-related HSS and MSS components are subject to the TARGETED QA program. These controls will be specifically 'targeted' to the critical attributes that resulted in the component being categorized as HSS or MSS. Components under these controls will remain non-safety-related, but the enhanced treatments will be appropriately applied to give additional confidence that the component will be able to perform its HSS/MSS function when demanded.

These identified processes provide reasonable confidence that HSS and MSS components will be able to perform their risk significant functions. The validation of functionality of HSS and MSS SSCs (safety-related SSCs for which existing special treatment does not provide the applicable level of confidence and non-safety-related SSCs) will consist of a documented technical evaluation under the corrective action program to determine what enhanced treatment, if any, is warranted for these SSCs to provide reasonable confidence that the applicable risk significant functions will be satisfied. The performance of these SSCs will be monitored to provide reasonable confidence of their ongoing capability to perform their risk significant functions. The design control process will be applied to facility changes affecting the risk-significant functions of these SSCs.

13.7.3.3 Normal Commercial and Industrial Practices for Safety-Related LSS and NRS Components. A description of STP's commercial practices is provided below. The purpose of applying these practices to safety-related LSS and NRS SSCs is to provide STP with reasonable confidence that these SSCs will maintain their functionality under design-basis conditions.

In lieu of any of these commercial practices, the associated special treatment requirements of NRC regulations may be applied to safety-related LSS and NRS components.

13.7.3.3.1 Design Control Process. The Station's Design Control Program is used for safety-related SSCs, including safety-related LSS and NRS SSCs. The Design Control Program complies with 10 CFR Part 50, Appendix B, and is described in the Operations Quality Assurance Plan (OQAP). Changes in the design functions of safety-related LSS and NRS SSCs or the conditions under which the intended functions are required to be performed, as described in the FSAR, will be controlled by following the design control process satisfying 10 CFR Part 50, Appendix B, and other regulatory requirements that may be applicable, such as 10 CFR 50.59.

13.7.3.3.2 Procurement Process. The purpose of the procurement process for safety-related LSS and NRS SSCs is to procure replacement SSCs that satisfy the design inputs and assumptions to support STP's determination that these SSCs will be capable of performing their safety-related functions under design-basis conditions. Technical requirements (including applicable design basis environmental and seismic conditions) for items to be procured include the design inputs and assumptions for the item. As described below, one or more of the following methods will provide a sufficient basis to determine that the procured item can perform its safety-related function under design basis conditions, including applicable design basis environmental (temperature and pressure, humidity, chemical effects, radiation, aging, submergence, and synergistic effects) and seismic (earthquake motion, as described in the design bases, including seismic inputs and design load combinations) conditions:

- Vendor Documentation - Vendor documentation could be used when the performance characteristics for the item, as specified in vendor documentation (e.g., catalog information, certificate of conformance), satisfy the SSC's design requirements. If the vendor documentation does not contain this level of detail, then the design requirements could be provided in the procurement specifications. The vendor's acceptance of the stated design specifications provides sufficient confidence that the replacement safety-related LSS or NRS SSC would be capable of performing its safety-related functions under design basis conditions. Differences constituting a design change will be documented and processed under the STP design control process.

- Equivalency Evaluation - An equivalency evaluation could be used when it is sufficient to determine that the procured item is equivalent to the item being replaced (e.g., a like-for-like replacement).

- Technical Evaluation - For minor differences, a technical evaluation could be performed to compare the differences between the procured item and the design requirements of the item being replaced and determines that differences in areas such as material, size, shape, stressors, aging mechanisms, and functional capabilities would not adversely affect the ability to perform the safety-related functions of the SSC under design basis conditions. Differences constituting a design change will be documented and processed under the STP design control process.

- Technical Analysis - In cases involving substantial differences between the procured item and the design requirements of the item being replaced, a technical analysis could be performed to determine that the procured item can perform its safety-related function under design basis conditions. The technical analysis would be based on one or more engineering methods that include, as necessary, calculations, analyses and evaluations by multiple disciplines, test data, or operating experience to support functionality of the SSC over its expected life. Where the differences are determined to require a design change, STP will follow the design control process for safety-related SSCs.

- Testing - Testing under simulated design basis conditions could be performed on the component. Margins and documentation specified in NRC regulations would not be required in these tests, since the components are LSS/NRS and do not warrant this additional confidence.

Documentation of the implementation of these methods is maintained. Additionally, documentation is maintained to identify the preventive maintenance needed to preserve the capability of the procured item to perform its safety-related function under applicable design basis environmental and seismic conditions for its expected life.

In the procurement process, STP uses standards required by the State of Texas and national consensus commercial standards used at STP for the procurement of SSCs consistent with STP's normal commercial and industrial practices. STP does not need to itemize the standards in use at STP or to perform an evaluation of all national consensus standards.

The procurement program provides for the identification and implementation of special handling and storage requirements to ensure that the item is not damaged or degraded during shipment to the site or during storage on site. These handling and storage requirements consider available recommendations from the vendor. STP may use an alternative to these recommendations if there is a technical basis that supports the functionality of the safety-related LSS and NRS SSCs. The basis does not need to be documented.

At the time of receipt, the received item is inspected to ensure that the item was not damaged in the process of shipping, and that the item received is the item ordered.

13.7.3.3.3 Installation Process. The purpose of the installation process for safety-related LSS and NRS SSCs is to achieve proper installation and testing of replacement SSCs to support STP's determination that these SSCs will be capable of performing their safety-related functions under design-basis conditions.

In the installation process, STP uses standards required by the State of Texas and national consensus commercial standards used at STP for the installation of SSCs consistent with STP's normal commercial and industrial practices. STP does not need to itemize the standards in use at STP or to perform an evaluation of all national consensus standards.

Post-installation testing will be performed to the extent necessary to provide STP with reasonable confidence that the installed SSC will perform its safety function. The test verifies that the SSC is operating within expected parameters and is functional. The testing may necessitate that the SSC be placed in service to validate the acceptance of its performance. Testing is not necessarily performed under design basis conditions.

13.7.3.3.4 Maintenance Process. The purpose of the maintenance process for safety-related LSS and NRS SSCs is to establish the scope, frequency, and detail of maintenance activities necessary to support STP's determination that these SSCs will remain capable of performing their safety-related functions under design-basis conditions. Preventive maintenance tasks are developed for active structures, systems, or components factoring in vendor recommendations. STP may use an alternative to these recommendations if there is a technical basis that supports the functionality of the safety-related LSS and NRS SSCs. For an SSC in service beyond its designed life, STP will have a technical basis to determine that the SSC will remain capable of performing its safety-related function(s). These bases, while documented, do not need to be retained as quality records.

The frequency and scope of predictive maintenance actions are established and documented considering vendor recommendations, environmental operating conditions, safety significance, and operating performance history. STP may deviate from vendor recommendations where a technical basis supports the functionality of the safety-related LSS and NRS SSCs. Such deviations are not required to be documented.

When an SSC deficiency is identified, it is documented and tracked through the Corrective Action Program. The deficiency is evaluated to determine the corrective maintenance to be performed.

Following maintenance activities that affect the capability of a component to perform its safety-related function, post maintenance testing is performed to the extent necessary to provide reasonable confidence that the SSC is performing within expected parameters.

In the maintenance process, STP uses standards required by the State of Texas and national consensus commercial standards used at STP for the maintenance of SSCs consistent with STP's normal commercial and industrial practices. STP does not need to itemize the standards in use at STP or to perform an evaluation of all national consensus standards.

13.7.3.3.5 Inspection, Test, and Surveillance Process. The purpose of the inspection, test, and surveillance process for safety-related LSS and NRS SSCs is to obtain data or information that allows evaluation of operating characteristics to support STP's determination that these SSCs will remain capable of performing their safety-related functions under design-basis conditions throughout the service life of the SSC. The Station's inspection and test process is primarily addressed and implemented through the Maintenance process. When measuring and test equipment is found to be in error or defective, a determination is made of the functionality of the safety-related SSCs that were checked using that equipment. As stated above, the Maintenance process addresses inspections and tests through corrective, preventive, and predictive maintenance activities. These activities factor in vendor recommendations into the selected approach. STP may use an alternative to these recommendations if there is a technical basis that supports the functionality of the safety-related LSS and NRS SSCs. The basis does not need to be documented.

In the inspection, test, and surveillance process, STP uses standards required by the State of Texas and national consensus commercial standards used at STP for the inspection and testing of SSCs consistent with STP's normal commercial and industrial practices. STP does not need to itemize the standards in use at STP or to perform an evaluation of all national consensus standards.

13.7.3.3.6 Corrective Action Program. The Station's Corrective Action Program is used for safety-related (LSS and NRS as well as HSS and MSS SSCs) applications. The Corrective Action Program complies with 10 CFR Part 50 Appendix B, and is described in the OQAP.

13.7.3.3.7 Management and Oversight Process. The purpose of the management and oversight process for safety-related LSS and NRS SSCs is to control the implementation and to assess the effectiveness of the commercial practices to support STP's determination that these SSCs will remain capable of performing their safety-related functions under design-basis conditions. The Station's management and oversight process is accomplished through approved procedures and guidelines.

Procedures provide for the qualification, training, and certification of personnel. STP considers vendor recommendations in the training, qualification, and certification of personnel. STP may use an alternative to these recommendations if there is a basis for continued effective training of personnel. The basis does not need to be documented.

For qualification, training, and certification of personnel, STP uses standards required by the State of Texas and national consensus commercial standards used at STP consistent with STP's normal commercial and industrial practices. STP does not need to itemize the standards in use at STP or to perform an evaluation of all national consensus standards.

Documentation, reviews, and record retention requirements for completed work activities are governed by Station procedures.

Planned changes to, or elimination of, commitments described in the UFSAR or other licensing bases documentation that address issues identified in NRC generic communications (e.g., generic letters or bulletins), NRC orders, notices of violation, etc. related to safety-related LSS and NRS SSCs will be evaluated in accordance with an NRC-endorsed commitment change process.

13.7.3.3.8 Configuration Control Process. The Station's configuration control process is controlled through approved procedures and policies. The design control process ensures that the configuration of the Station is properly reflected in design documents and drawings.

13.7.4 Continuing Evaluations and Assessments

13.7.4.1 Performance Monitoring. STP has performance monitoring processes that include the following:

- Maintenance Rule Program - Specific performance criteria are identified at the plant, system, or train level. Regardless of their risk categorization, components that affect MSS or HSS functions will be monitored and assessed in accordance with plant, system and/or train performance criteria.

- Corrective Action Program - Condition reports document degraded equipment performance or conditions, including conditions identified as a result of operator rounds, system engineer walk-downs, and corrective maintenance activities.

- STP collects indicators from the performance of plant activities, such as corrective maintenance, installation of modifications, and conduct of testing.

13.7.4.2 Feedback and Corrective Action. STP has feedback and corrective action processes to ensure that equipment performance changes are evaluated for impact on the component risk categorization, the application of special treatment, and other corrective actions. At least once every two refueling outages, performance data is compiled for review, which is performed for each system that has been categorized in accordance with Section 13.7.2. Performance and reliability data are generally obtained from sources such as the Maintenance Rule Program and Operating Experience Review.

This process provides an appropriate level of assurance that any significant negative performance changes that are attributed to the relaxation of special treatment controls are addressed in a timely manner. Responsive actions may include the reinstatement of applicable controls up to and including the re-categorization of the component's risk significance, as appropriate.

13.7.4.3 Process for Assessing Aggregate Changes in Plant Risk. The designated individuals who implement the integrated decision-making process are responsible for assessing and approving the aggregate effect on plant risk for risk-informed applications.

The process used to access the aggregate change in plant risk associated with changes in special treatment for components is based on periodic updates to the station's PRA and the associated PRA risk ranking sensitivity studies.

13.7.5 Quality Assurance and Change Control for the Risk-Informed Process

13.7.5.1 Quality Assurance for the PRA Risk Categorization Process.

STP has a PRA configuration control program, which is structured to ensure that changes in plant design and equipment performance are reflected in the PRA as appropriate. The PRA configuration control process is controlled by procedures and guidelines that ensure proper control of changes to the models.

13.7.5.2 Regulatory Process for Controlling Changes. Changes affecting Section 13.7 will be controlled in accordance with the following provisions:

a. Changes to Section 13.7.2, "Component Categorization Process" may be made without prior NRC approval, unless the change would decrease the effectiveness of the process in identifying HSS and MSS components.

b. Changes to Section 13.7.3, "Treatment of Component Categories" may be made without prior NRC approval, unless the change would result in a reduction in the confidence of component functionality.

c. Changes to Section 13.7.4, "Continuing Evaluations and Assessments" may be made without prior NRC approval, unless the change would result in a decrease in effectiveness of the evaluations and assessments.

d. A report shall be submitted, as specified in 10 CFR 50.4, of changes made without prior NRC approval pursuant to these provisions. The report shall identify each change and describe the basis for the conclusion that the change does not involve a decrease in effectiveness or confidence as described above. The report shall be submitted within 60 days of the date of the change.

e. Changes to Sections 13.7.2, 13.7.3, and 13.7.4 that do not meet the criteria of Sections 13.7.5.2.a through c shall be submitted to the NRC for prior review and approval.

TABLE 13.7-1

EXEMPTIONS FROM SPECIAL TREATMENT REQUIREMENTS

Regulation	Scope of Exemption
10 CFR 21.3 – An exemption to exclude safety-related LSS and NRS components from the scope of the definition of "basic component."	The procurement, dedication, and reporting requirements in Part 21 are not applied to safety-related LSS and NRS components.
10 CFR 50.34(b)(10) and (11) – An exemption to the extent that it incorporates seismic qualification requirements in Part 100.	Refer to request for exemption from Part 100.
10 CFR 50.49(b) – An exemption to exclude LSS and NRS components from the scope of electric equipment important to safety for the purposes of environmental qualification of electrical equipment.	• The qualification documentation and files specified in Section 50.49 are not applicable to LSS and NRS components. • LSS and NRS components are not required to be maintained in a qualified condition under Section 50.49. • LSS and NRS components may be replaced with components that are not qualified under Section 50.49. • LSS and NRS components, as applicable under Section 50.49, are designed to function in the applicable design basis environment. Section 13.7.3.3 identifies the design and procurement controls that are applied to LSS and NRS components to achieve this requirement.
10 CFR 50.55a(g) – An exemption from the requirements of ASME Section XI, for repair and replacement of ASME Class 2 and 3 safety-related LSS and NRS components, subject to the provisions identified in the scope of exemption.	ASME Class 2 and 3 safety-related LSS and NRS components may be repaired or replaced with components that meet one of the following alternatives. The term 'item' below includes repairs, replacements, and fabrication and installation welds categorized as LSS or NRS : • The repair or replacement item will meet the technical (but not the administrative) requirements of the ASME Section XI Code and of the ASME Construction Code, as incorporated in Section XI. • The repair or replacement item will meet the technical and administrative requirements of other nationally-recognized Codes, Standards, or Specifications suitable for the item. Section 13.7.3.3 identifies the quality, design and procurement controls that are applied to safety-related LSS and NRS components that are repaired or replaced to provide reasonable confidence that their functionality is maintained.

TABLE 13.7-1 (Continued)

EXEMPTIONS FROM SPECIAL TREATMENT REQUIREMENTS

Regulation	Scope of Exemption
10 CFR 50.55a(f) – An exemption from meeting the requirements of ASME Section XI for testing of safety-related LSS and NRS components.	Safety-related LSS and NRS components are not in the scope of component-specific inservice testing requirements. Additionally, Section 13.7.3.3 identifies other controls that are applied to provide reasonable confidence that safety-related LSS and NRS component functionality is maintained.
10 CFR 50.55a(g) – An exemption from meeting the requirements of ASME Section XI for inservice inspection of safety-related LSS and NRS components, subject to the provisions in the Scope of Exemption.	Safety-related LSS and NRS components are not in the scope of inservice inspection requirements. Section 13.7.3.3 identifies controls that are applied to provide reasonable confidence that safety-related LSS and NRS component functionality is maintained.
10 CFR 50.55a(h) – An exemption to exclude safety-related LSS and NRS components from the scope of components required to meet sections 4.3 and 4.4 of IEEE 279.	Sections 4.3 and 4.4 of IEEE 279 do not apply to safety-related LSS and NRS components. The other requirements listed in IEEE 279, including functional and design requirements, are applicable. Additionally, Section 13.7.3.3 identifies other controls that are applied to provide reasonable confidence that safety-related LSS and NRS component functionality is maintained.
10 CFR 50.59(a)(1), (a)(2) and (b)(1) (pre-1999 version); 10 CFR 50.59(c)(1), (c)(2), and (d)(1) (2000 version) – An exemption from the requirement to perform a written evaluation of changes in special treatment requirements for LSS and NRS components. Also an exemption from the requirement to seek prior NRC approval for such changes to the extent that they fall within the listed criteria in 50.59.	STP is not required to perform 50.59 evaluations for changes in the special treatment requirements for LSS and NRS components, and is not required to seek prior NRC approval for those changes. The exemption is limited to changes in special treatment requirements for which the exemption has been granted.
10 CFR 50.65(b) – An exemption to exclude LSS and NRS components from the scope of SSCs covered by the Maintenance Rule (except for 10 CFR 50.65(a)(4)).	STP is required to monitor performance on a plant/system/train level, as applicable. Regardless of their risk categorization, components that affect MSS or HSS functions will be monitored and assessed in accordance with plant, system, and/or train performance criteria.

TABLE 13.7-1 (Continued)

EXEMPTIONS FROM SPECIAL TREATMENT REQUIREMENTS

Regulation	Scope of Exemption
10 CFR Part 50 Appendix B, Introduction – An exemption to exclude safety-related LSS and NRS components from the scope of safety-related SSCs covered by Appendix B (except for Criterion III pertaining to Design Control and Criteria XV and XVI governing non-conformances and corrective actions).	• Safety-related LSS and NRS components are not required to satisfy the QA requirements in Appendix B, except for design control, control of nonconformances, and corrective action. • Section 13.7.3.3 identifies other controls that are applied to provide reasonable confidence that safety-related LSS and NRS component functionality is maintained.
10CFR Part 50, Appendix J, B.III – An exemption to exclude safety-related LSS and NRS components, subject to the additional limitations listed under Scope of Exemption, from the scope of components requiring local leak rate tests and containment isolation valve leak rate tests.	• Local leak rate tests of LSS containment isolation valves and other safety-related LSS or NRS components are not required. With respect to LSS containment isolation valves, this exemption only applies to valves that satisfy one or more of the following criteria: - The valve is required to be open under accident conditions to prevent or mitigate core damage events. - The valve is normally closed and in a physically closed, water-filled system. - The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and that is not connected to the reactor coolant pressure boundary. - The valve is in a closed system whose piping pressure rating exceeds the containment design pressure rating, and is connected to the reactor coolant pressure boundary. The process line between the containment isolation valve and the reactor coolant pressure boundary is non-nuclear safety. - The valve size is 1 inch NPS or less. - Cumulative limits for containment leakage are based upon the tested components, with the assumption that the exempted components contribute zero leakage. • Section 13.7.3.3 identifies controls that are applied to provide reasonable confidence that safety-related LSS and NRS component functionality is maintained.
10 CFR Part 100, Appendix A.VI(a)(1) and (2) – An exemption	• LSS and NRS components are not required to be maintained in a qualified condition under Part 100.

TABLE 13.7-1 (Continued)

EXEMPTIONS FROM SPECIAL TREATMENT REQUIREMENTS

Regulation	Scope of Exemption
to exclude safety-related LSS and NRS components from the scope of SSCs covered by these sections, to the extent that these sections require testing and specific types of analyses to demonstrate that SSCs are designed to withstand the safe shutdown earthquake and operating basis earthquake.	• LSS and NRS components may be replaced with components that are not qualified under Part 100. • LSS and NRS components, as applicable under Part 100, are designed to withstand the effects of design basis seismic events without loss of capability to perform their safety function. Section 13.7.3.3 identifies the design and procurement controls that are applied to LSS and NRS components to achieve this requirement.

CN-3235

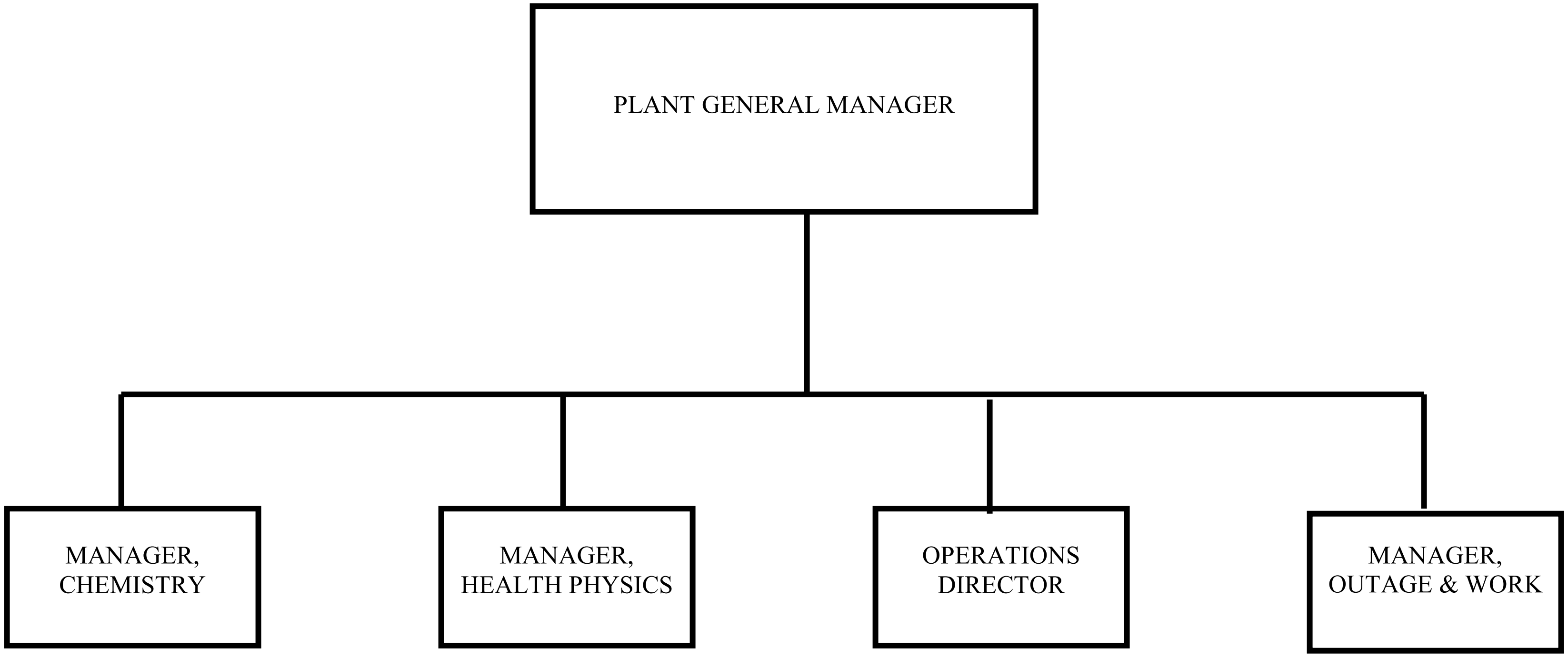

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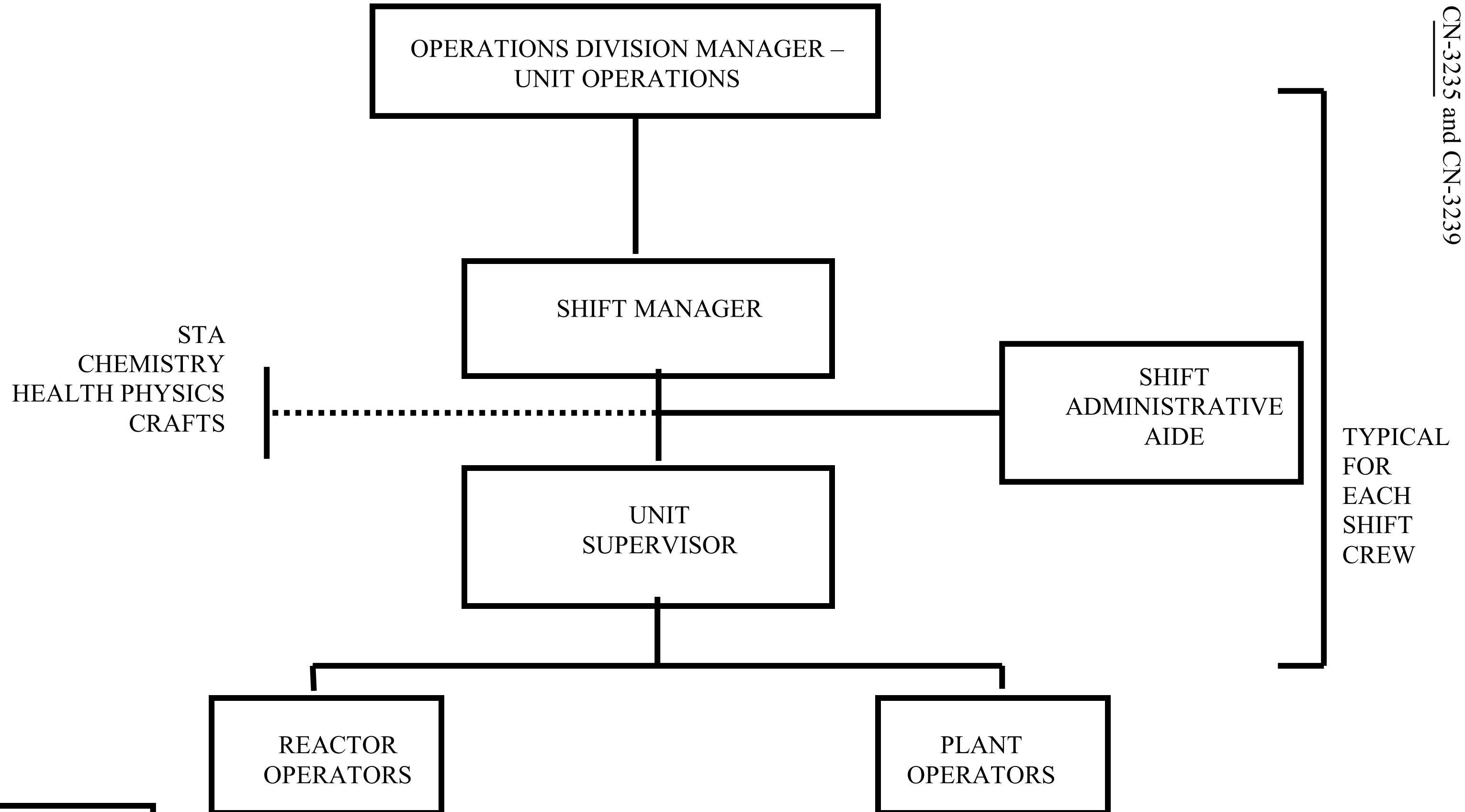
FIGURE 13.1-1 Revision 19

CN-3235 and CN-3239



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SHIFT OPERATION
ORGANIZATION

FIGURE 13.1-2 Revision 19