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TABLE 4.1-2a

CHEMICAL ANALYSES IN WEIGHT PERCENT
REACTOR VESSEL SURVEILLANCE MATERIAL

Element	Intermediate Shell		Lower Shell	
	Unit 3	Unit 4	Unit 3	Unit 4
C	0.20	0.22	0.20	0.21
Mn	0.64	0.67	0.61	0.67
P	0.010	0.010	0.010	0.011
S	0.010	0.009	0.008	0.009
Si	0.26	0.20	0.20	0.23
Ni	0.70	0.71	0.67	0.70
Cr	0.40	0.33	0.38	0.31
V	0.02	0.002	0.02	0.001
Mo	0.62	0.56	0.58	0.56
Co	0.011	0.017	0.015	0.015
Cu	0.058	0.054	0.079	0.056
Zr	*0.001	0.005	*0.001	0.004
Sn	0.010	0.008	0.008	0.008
Ti	*0.001	*0.001	*0.001	*0.001
Sb	*0.001		*0.001	
Zn	0.001	*0.001	0.001	*0.001
As	*0.005	0.004	*0.005	0.005
B	*0.003	*0.003	*0.003	*0.003
Al	0.005	0.008	0.005	0.008
N^2	0.003	0.001	0.003	0.002
Nb		0.002		0.001
W		*0.001		*0.001
Pb		*0.001		0.001
Ta		0.003		0.002

* Not detected. The number indicates the minimum limit of detection.

TABLE 4.1-3

PRESSURIZER AND PRESSURIZER RELIEF TANK DESIGN DATA

Pressurizer	
Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3107
Design/Operating Temperature °F	680/653
Water Volume, Full Power, ft^3 *	780
Steam Volume, Full Power, ft^3	520
Surge Line Nozzle Diameter, in./Pipe Schedule	14/Sch 140
Shell ID, in./Minimum Shell Thickness, in.	84/4.1
Mimimum Clad Thickness, in.	0.188
Electric Heaters Capacity, kw (total)**	1300(Design)
Heatup rate of Pressurizer using Heaters only, °F/hr	55 (approximately with Design heater capacity)
Power Relief Valves: #455C & 456	
Number	2
Set Pressure (open), psig	
i) Normal operation	2335
ii) OMS Actuation during Heatup or Cooldown	
a) RCS ≤ 285°F	415 ±15
b) RCS > 285°F	Setpoint increases step-wise to 2335 psig as temperature increases to 750°F (See Table 4.1-1)
Capacity, lb/hr saturated steam/valve	179,000
Safety Valves	
Number	3
Set Pressure, psig	2485 ±1% (as left) +2%/-3% (as found)
Capacity, lb/hr saturated steam/valve	293,330
Pressurizer Relief Tank	
Design pressure, psig	100
Rupture disc release pressure, psig	100
Design temperature, °F	340
Normal water temperature, °F	120
Total volume, ft^3	1300
Rupture disc relief capacity, lb/hr	900,000

* 60% of net internal volume (maximum calculated power)

** Original as-built design. Thermal uprate analysis uses 1000 kw minimum pressurizer heater capacity.

shows that failure could occur if vertical reinforcing were not provided. In fact, the maximum allowable vertical averaged tensile stress according to Taylor's interaction curve is

$$\frac{f_a}{f'_c} = 0.03$$

therefore, $f_a = +150$ psi. For this reason, special anchorage zone reinforcing is used in addition to that required by the loading cases. Such special reinforcing is based on the following considerations:

1. Full scale load tests of the anchorage on the same concrete mix used in the structure and review of prior uses of the anchorage.

2. The post-tensioning supplier's recommendations of anchorage reinforcing requirements.

3. Review of the final details of the combined reinforcing by the consulting firm of T. Y. Lin, Kulka, Yang and Associate.

For typical detailed Analysis, see Topical Report B-Top-2 dated October 1969, submitted in connection with Docket No. 50-255, a NON-PROPRIETARY report.

(b) Earthquake or Wind Loading

The stresses in the structure for the earthquake loading conditions exceed the stresses for design tornado or wind. The earthquake analysis is conducted in the following manner:

The loads on the containment structure caused by earthquake are determined by a dynamic analysis of the structure. The dynamic analysis is made on an idealized structure of lumped masses and weightless elastic columns acting as springs.

The analysis is performed in two stages; the determination of natural frequencies of the structure and its mode shapes, and the response of these modes to the earthquake by the spectrum response. For the supported equipment, piping, etc. a time history technique is used to develop the floor response spectrum curves, and the supported elements are then analyzed by the response spectrum method as discussed in Appendix 5A, Section 5A-2.0.

The natural frequencies and mode shapes are computed using the matrix equation of motion shown below for a lumped mass system. Matrix interation was performed by use of a digital computer program to yield the natural frequencies and mode shapes. The form of the equation is:

$$(K) \cdot (\Delta) = \omega^2 \cdot (M) \cdot (\Delta)$$

K = Matrix of stiffness coefficients including the combined effects of shear, flexure, rotation and horizontal translation.
M = Matrix of lumped masses
Δ = Matrix of mode shapes
ω = Angular natural frequency of vibration

The results of this computation are the several values of ω_n and mode shapes Δn for n = 1, 2, 3, ---m where m is the number of degrees of freedom (i.e., lumped masses) assumed in the idealized structure.

To obtain the loads on the containment structure the response of each mode of vibration to the design earthquake is computed by the response spectrum technique as follows:

newer structures, wind loads are as required by the edition of the South Florida Building Code applicable at the time of design. Shape Factors are applied in accordance with Reference 5A-4, or as required by the South Florida Building Code applicable at the time of design. No tornado loads are considered.

5A-1.4.2 Turkey Point Fossil Units 1 and 2 Chimney Design Requirements

The Fossil Unit 1 & 2 chimneys, located directly north of Unit 3, do not perform any safety related functions, or directly protect safety related equipment. However, failure of these structures has the potential of adversely affecting safety related systems. Accordingly, these structures have been designed to not fail and cause an adverse interaction with any safety related systems, when subjected to the Class I seismic loads (0.15 g) and wind loads (145 mph hurricane and 225 mph tornado) described in Sections 5A-1.3.4 and 5A-1.3.5 of this appendix.

5A-1.5 Miscellaneous Loads for Structures, Systems and Equipment

The units are designed for an outdoor temperature range of +30°F to +95°F. No ice or snow loads are considered in the design of the various structures and equipment.

External flood protection is described in Appendix 5G.

5A-2.0 METHOD OF SEISMIC ANALYSIS

5A-2.1 Structures

The methods for seismic analysis of the containment and control building structures are described in Section 5.1.3.2.

5A-2.2 Response Spectra

Response spectra curves for floors at grade and for the containment basemat were developed based on the El Centro, California, earthquake. These curves are shown in Figures 5A-1 for the design basis earthquake event (E), and Figure 5A-2 for the maximum earthquake event (E'). For class I piping, floor response spectra for the connecting points are developed. Additionally, response spectra curves are also generated for the control building. The analysis methodology is similar to the technique described in Section 5.1.3.2(b). (Reference 5A-3)

5A-2.3 Seismic Class I Piping Analysis

Seismic Class I piping systems are typically analyzed as mathematical models consisting of lumped masses connected by elastic members. The distance from the pipe axis to the center of gravity of the valve and operator is considered, with the mass of the valve and operator, for all motor, air, or gear operated valves. When necessary for the integrity of the piping, valve, or operation, the valve structure is externally supported. The stiffness matrix for the pipe is developed to include the effects of torsional, bending, shear and axial deformations as well as change in flexibility due to curved members and internal pressure. Flexibility factors are calculated in accordance with USAS B31.1. System natural frequencies and mode shapes for all significant modes of vibration are then determined using equations of motion, and spectral accelerations as determined from the response spectra applied.

The following equations are successively used to determine the response for each mode, maximum displacement for each mode, and the total displacement for each mass point:

(1) $$Y_n(\max) = \frac{R_n Sa_n D}{M_n \omega_n^2}$$

(2) $$V_{in} = \varphi_{in} Y_n(\max)$$

(3) $$V_i = \sqrt{\Sigma V_{in}^{\ 2}}$$

where:

$Y_n(max)$ = response of the n^{th} mode
R_n = participation factor for the n^{th} mode = $\Sigma M_i \phi_{in}$
M_i = mass i
ϕ_{in} = mode shape i for n^{th} mode
Sa_n = spectral acceleration for the n^{th} mode
D = earthquake direction matrix
M_n = generalized mass matrix for the n^{th} mode = $\Sigma M_i \phi^2_{in}$
ω_n = angular frequency of the n^{th} mode
V_{in} = maximum displacement of mass i for mode n
V_i = maximum displacement of mass i due to all modes calculated

9.5.3 SYSTEM EVALUATION

Underwater transfer of spent fuel provides essential ease and corresponding safety in handling operations. Water is an effective, economic and transparent radiation shield and a reliable cooling medium for removal of decay heat.

Basic provisions to ensure the safety of refueling operations are:

a) Gamma radiation levels in the containment, control room and fuel storage areas are continuously monitored (see Section 11.2.3). These monitors provide an audible alarm at the initiating detector indicating an unsafe condition. Continuous monitoring of reactor neutron flux provides immediate indication and alarm in the control room of an abnormal core flux level.

b) Containment integrity is maintained when core alterations or movement of irradiated fuel occurs inside the containment.

c) Whenever any fuel is being added to the reactor core or is being relocated, a reciprocal curve of source neutron multiplication is recorded to verify the subcriticality of the core.

Incident Protection

Direct communication between the control room and the refueling cavity manipulator crane is required whenever changes in core geometry which affect criticality are taking place. This provision allows the control room operator to inform the manipulator crane operator of any impending unsafe conditions detected from the control board indicators during fuel movement.

Malfunction Analysis

An analysis is presented in Section 14 concerning damage to one complete outer row of fuel elements in an assembly, assumed as a conservative limit for evaluating environmental consequences of a fuel handling incident.

9.5.4 TEST AND INSPECTION CAPABILITY

Upon completion of core loading and installation of the reactor vessel head, certain mechanical and electrical tests can be performed prior to initial criticality. The electrical wiring for the rod drive circuits, the rod position indicators, the reactor trip circuits, the in-core thermocouples and the reactor vessel head water temperature thermocouples can be tested at the time of installation. The tests can be repeated on these electrical items before initial operation.

9.5.5 REFERENCE

1. Turkey Point Unit 4 Plant Change Modification (PC/M) 95-066, "Turkey Point Unit 4 Cycle 16 Reload," Revision 2, dated March 6, 1996.

Appendix A Guidelines	Plant Conformance	Alternatives	Remarks
G.4 Materials Containing Radioactivity Materials that collect and contain radioactivity such as spent ion exchange, resins charcoal filters, and HEPA filters should be stored in closed metal tanks or containers that are located in areas free from ignition sources or combustibles. These materials should be protected from exposure to fires in adjacent areas as well. Consideration should be given to requirements for removal of isotopic decay heat from entrained radioactive materials.	Materials containing or collecting radioactivity are stored in closed metal containers in areas free of ignition sources or combustibles. Barriers and separation are provided to preclude exposure to fire in adjacent areas. Requirements for control of decay heat are developed for specific storage materials, as required.		

2.5 CONFORMANCE TO 10 CFR PART 50 APPENDIX R REQUIREMENTS

The information which follows is a lineup of the Turkey Point Units 3 and 4 designs against the requirements of Appendix R to 10 CFR Part 50. Also see the lineup against BTP Appendix A presented in Section 2.4 of this Appendix.

Appendix R requirements are given in the first (left-hand) column of the following tabulations, retaining the numbering sequence of Appendix R. Information on various aspects of the Turkey Point Units 3 and 4 Fire Protection Program is given in the second column as necessary to demonstrate conformance to the Appendix R Requirements, or in the third column to describe alternative approaches. The fourth column provides supplemental information as appropriate.

Based on the criteria established in 10 CFR Part 50.48, Turkey Point Units 3 and 4 are required to conform only to Sections III.G, III.J, and III.0 of Appendix R. Additional Sections requiring conformance as a result of prior NRC review and acceptance of Turkey Point Units 3 and 4 design with respect to BTP APCSB 9.5-1 Appendix A are III.A, III.H, III.I and III.L. All other Sections of Appendix R are not applicable to Turkey Point Units 3 and 4.

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11.4 Radiological Administrative Controls

The following programs shall be established, implemented, and maintained:

11.4.1 In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(1) Training of personnel,

(2) Procedures for monitoring, and

(3) Provisions for maintenance of sampling and analysis equipment.

11.4.2 Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

2. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3. Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

11.4.3 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

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12. CONDUCT OF OPERATIONS

12.1 Organization and Responsibility

This section covered the positions and personnel at the time of initial plant startup and operation. This information can be found in the original docketed FSAR and this is also addressed in the plant operating license (Technical Specifications).

12.2 Training

This section covered the training program at the time of initial plant startup and operation. This information can be found in the original docketed FSAR and this is also addressed in the Technical Specifications.

12.3 Procedures

The operating procedures for startup, normal operations, and anticipated emergency operating conditions is addressed in the original docketed FSAR and current requirements indicated in the Technical Specifications and in this chapter. The Emergency Plan in effect for Turkey Point is issued as a separate document.

12.4 Records

The procedure for maintaining plant operating, maintenance, QA, personnel, training, and instrumentation and control record is addressed in the original docketed FSAR and current requirements indicated in the Technical Specifications and in this chapter.

12.5 Administrative Control

The necessary administrative procedures are addressed in the original docketed FSAR and current requirements indicated in the Technical specifications and in this chapter.

12.6 Plant Security Plan

Turkey Point maintains a Plant Security Plan and is issued as a separate document.

The FPL Quality Assurance Program is described by FPL Topical Quality Assurance Report FPLTQAR 1-76A. This corporate level document is supplemented by the following sections of Chapter 12 which contain plant-specific details of the FPL QA program related to Administrative Controls. Sections 12.7 through 12.10 were originally located in the Technical Specifications and have been relocated to the UFSAR by Technical Specifications Amendment No. 201/195 (NRC Safety Evaluation Report related to Amendment No. 201/195, dated October 6, 1999). These sections are not only subject to the regulatory requirements of 10 CFR 50.54(a) with respect to changes to approved QA program description, but also subject to the regulatory requirements of 10 CFR 50.59.

12.7 Review and Audit

12.7.1 Plant Nuclear Safety Committee (PNSC)

12.7.1.1 Function

The PNSC shall function to advise the Plant General Manager on all matters related to nuclear safety.

12.7.1.2 Composition

The PNSC shall have a minimum of nine voting members and be composed of individuals from each of the following disciplines:

Operations	Technical Support
Maintenance	Licensing
Health Physics	Quality Assurance/Control
Reactor Engineering	Instrument and Control
Protection Services	

The PNSC Chairman and Vice-Chairman shall be appointed in writing from among the members by the Plant General Manager.

The members, according to individual job titles, shall meet the requirements as described in Sections 4.2, 4.3, or 4.4, of the ANSI N-18.1-1971.

12.7.1.3 Alternates

All alternate members shall be appointed in writing by the Plant General Manager to serve on a temporary basis; however, no more than two alternates shall participate as members in PNSC activities at any one time.

12.7.1.4 Meeting Frequency

The PNSC shall meet at least once per calendar month and as convened by the PNSC Chairman or his designated alternate.

12.7.1.5 Quorum

The quorum of the PNSC necessary for the performance of the PNSC responsibility and authority provisions of Section 12.7.1 shall consist of the Chairman or Vice Chairman and four members including alternates.

12.7.1.6 Responsibilities

The PNSC shall be responsible for:

a. Review of all safety-related plant administrative procedures and changes thereto.

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to Appendix "A" Technical Specifications;

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety;

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the President-Nuclear Division and to the Chairman of the Company Nuclear Review Board;

f. Review of all REPORTABLE EVENTS

g. Review of reports of significant operating abnormalities or deviations from normal and expected performance of plant equipment or systems that affect nuclear safety.

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Plant General Manager or the Chairman of the Company Nuclear Review Board;

i. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL;

j. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the President-Nuclear Division and to the Chairman of the Company Nuclear Review Board.

k. Review of the Fire Protection Program and implementing procedures and the submittal of recommended changes to the Company Nuclear Review Board.

l. Review of the Diesel Fuel Oil Testing Program and implementing procedures.

12.7.1.7 The PNSC shall:

a. Recommend in writing to the Plant General Manager approval or disapproval of items considered under Subsections 12.7.1.6a. through d. prior to their implementation and items considered under Subsections 12.7.1.6i through k.

b. Provide written notification within 24 hours to the Plant General Manager, President-Nuclear Division and the Company Nuclear Review Board of disagreement between the PNSC and the Plant General Manager; however, the Plant General Manager shall have responsibility for resolution of such disagreements pursuant to Technical Specification 6.1.1.

12.7.1.8 Records

The PNSC shall maintain written minutes of each PNSC meeting that, at a minimum, document the results of all PNSC activities performed under the responsibility provisions of Section 12.7.1. Copies shall be provided to the President-Nuclear Division and the Company Nuclear Review Board.

12.7.2 Company Nuclear Review Board (CNRB)

12.7.2.1 Function

The CNRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,
b. Nuclear engineering,
c. Chemistry and radiochemistry,
d. Metallurgy,
e. Instrumentation and control,
f. Radiological safety,
g. Mechanical and electrical engineering, and
h. Quality assurance practices.

The CNRB shall report to and advise the President-Nuclear Division on those areas of responsibility specified in Sections 12.7.2.7 and 12.7.2.8.

12.7.2.2 Composition

The President-Nuclear Division shall appoint, in writing, a minimum of five members to the CNRB and shall designate from this membership, in writing, a Chairman. The membership shall function to provide independent review and audit in the areas listed in Section 12.7.2.1. The Chairman shall meet the requirements of ANSI/ANS - 3.1 - 1987, Section 4.7.1. The members of the CNRB shall meet the educational requirements of ANSI/ANS - 3.1 - 1987, Section 4.7.2, and have at least 5 years of professional level experience in one or more of the fields listed in Section 12.7.2.1. CNRB members who do not possess the educational requirements of ANSI/ANS - 3.1 - 1987, Section 4.7.2 (up to a maximum of two members) shall be evaluated, and have their membership approved and documented, in writing, on a case-by-case basis by the President-Nuclear Division, considering the alternatives to the educational requirements of ANSI/ANS - 3.1 - 1987, Sections 4.1.1 and 4.1.2.

12.7.2.3 Alternates

All alternate members shall be appointed in writing by the CNRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in CNRB activities at any one time.

12.7.2.4 Consultants

Consultants shall be utilized as determined by the CNRB Chairman to provide expert advice to the CNRB.

12.7.2.5 Meeting Frequency

The CNRB shall meet at least once per 6 months and as convened by the CNRB chairman or his designated alternate.

12.7.2.6 Quorum

The quorum of the CNRB necessary for the performance of the CNRB review and audit functions of Section 12.7.2 shall consist of the Chairman or his designated alternate and at least four CNRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

12.7.2.7 Review

The CNRB shall be responsible for the review of:

a. The safety evaluations for: (1) changes to procedures, equipment, or systems; and (2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. All reportable events;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the PNSC.

12.7.2.8 Audits

Audits of unit activities shall be performed under the cognizance of the CNRB. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions;

b. The performance, training, and qualifications of the entire facility staff;

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or method of operation that affect nuclear safety;

d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50;

e. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel;

f. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

g. The Radiological Environmental Monitoring Program and the results thereof;

h. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures;

i. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes;

j. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring;

k. The Diesel Fuel Oil Testing Program and implementing procedures; and

l. Any other area of unit operation considered appropriate by the CNRB or the President-Nuclear Division.

12.7.2.9 <u>Records</u>

Records of CNRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each CNRB meeting shall be prepared, approved, and forwarded to the President-Nuclear Division within 14 days following each meeting;

b. Reports of reviews encompassed by Section 12.7.2.7 shall be prepared, approved, and forwarded to the President-Nuclear Division within 14 days following completion of the review; and

c. Audit reports encompassed by Section 12.7.2.8 shall be forwarded to the President-Nuclear Division and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

12.7.3 <u>Technical Review and Control</u>

12.7.3.1 <u>Activities</u>

Activities that affect nuclear safety shall be conducted as follows:

a. Procedures required by Technical Specification 6.8, and other procedures that affect nuclear safety, and changes thereto, shall be prepared, reviewed, and approved. Each such procedure, or change thereto, shall be reviewed by an individual/group other than the individual/group who prepared the procedure, or change thereto, but who may be from the same organization as the individual/group who prepared the procedure, or change thereto. Procedures other than plant administrative procedures shall be approved by the Plant General Manager, Operations Manager, or the head of the department assigned responsibility for those procedures prior to implementation. The Plant General Manager shall approve plant administrative procedures and emergency plan implementing procedures. Security Plan and the implementing procedures shall be approved by Protection Services Manager prior to implementation. Changes to procedures that may involve a change to the intent of the original procedures shall be approved by the individual authorized to approve the procedure prior to implementation of the change.

b. Individuals responsible for reviews performed in accordance with Subsection 12.7.3.1 (a) shall be members of the plant staff previously designated by the Plant General Manager and meet or exceed the minimum qualifications of ANSI N18.1-1971, Sections 4.2, 4.3.1, 4.4 and 4.6.1.

c. Each review shall include a determination of whether or not additional, cross-disciplinary review is necessary. If deemed necessary, such review shall be performed by qualified personnel of the appropriate discipline.

d. Each review will include a determination of whether or not an unreviewed safety question is involved.

12.7.3.2 Records of the above activities shall be provided to the Plant General Manager, PNSC, and/or the CNRB as necessary for required reviews.

12.8 Procedures and Programs

12.8.1 Each procedure listed in Technical Specification 6.8.1, and changes thereto, except the Quality Control Program for environmental monitoring, shall be reviewed and approved prior to implementation and reviewed periodically as set forth in Section 12.7.3 and administrative procedures.

12.8.2 Temporary changes to procedures listed in Technical Specification 6.8.1 may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license on the unit affected; and

c. The change is documented, reviewed in accordance with Section 12.7.3 and approved by the Plant General Manager or the department head of the responsible department within 14 days of implementation.

12.9 Record Retention

12.9.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

12.9.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety;

c. All reportable events;

d. Records of surveillance activities, inspections, and calibrations required by the Technical Specifications;

e. Records of changes made to the procedures required by Technical Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

12.9.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories;

c. Records of facility radiation and contamination surveys;

d. Records of radiation exposure for all individuals entering radiation control areas;

e. Records of gaseous and liquid radioactive material released to the environs;

f. Records of transient or operational cycles for those unit components identified in Technical Specification Table 5.7-1;

g. Records of reactor tests and experiments;

h. Records of training and qualification for current members of the facility staff;

i. Records of inservice inspections performed pursuant to the Technical Specifications;

j. Records of quality assurance activities required for the duration of the unit Operating License by the Quality Assurance Manual;

k. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

l. Records of meetings of the PNSC and the CNRB;

m. Records of the service lives of all hydraulic and mechanical snubbers required by Technical Specification 3.7.6 including the date at which the service life commences and associated installation and maintenance records;

n. Records of secondary water sampling and water quality; and

o. Annual Radiological Environmental Operating Reports and records of analyses transmitted to the licensee which are used to prepare the Annual Radiological Environmental Operating Report.

p. Records for Environmental Qualification which are covered under the provisions of 10 CFR 50.49.

q. Records of reviews performed for changes made to the Offsite Dose Calculation Manual and the Process Control Program

12.10 <u>Process Control Program (PCP)</u>

12.10.1 The Process Control Program (PCP) shall contain the current formulas, Sampling, analyses, test and determinations to be make to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

12.10.2 Licensee-initiated changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Section 12.9.3.q. This documentation shall contain:

1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the PNSC and the approval of the Plant Manager.