5.0 DESIGN FEATURES

6.0 ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
14. Intermediate Range Neutron Flux	2	1
15. Intermediate Range Neutron Flux Rate	2	1
16. Containment Isolation Valve Position*	2/Penetration	1/Penetration
17. Containment Enclosure Negative Pressure	2	1
18. Condensate Storage Tank Water Level**	2	1
19. Reactor Vessel Level Indication System	2	1
20. Containment Hydrogen Concentration	2	1
21. Containment Sump Isolation Valve Position***	2 (1 per valve)	1

*Applies to penetrations with 2 active valves in series. These valves are moved to the closed position by automatic signals.

**Calculated on basis of pressure sensed at suction to the Emergency Feedwater Pumps.

***Applies to CBS-V8 and CBS-V14 open indication on UL indicators.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result for first sample
			C-3	Perform Action for C-3 result of first sample	N.A	N.A.
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G.	All other S.G.s are C-1	None	N.A	N.A.
		Notification to NRC pursuant to §50.72 (b)(3) of 10CFR Part 50	Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N.A	N.A.
			Additional S.G. in C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b)(3) of 10CFR Part 50	N.A.	N.A.

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

6.1 RESPONSIBILITY

6.1.1 The Station Director shall be responsible for overall station operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Manager (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Site Vice President shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions for departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR and updated in accordance with the requirements of 10 CFR 50.71.

b. The Station Director shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Site Vice President shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION[1]

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, or 4	MODE 5 or 6
SM[2,4]	1	1
SRO[4]	1	None
RO	2	1
NSO	2	1
STA	1[3]	None

SM - Shift Manager with a Senior Reactor Operator license on Unit 1
SRO - Individual with a Senior Reactor Operator license on Unit 1
RO - Individual with an Operator license on Unit 1
NSO - Nuclear Systems Operator
STA - Shift Technical Advisor

TABLE NOTATIONS

(1) The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewperson being late or absent.

(2) During any absence of the Shift Manager from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Manager from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

(3) The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Manager or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.

(4) While the unit is in MODE 1, 2, 3 or 4, a licensed senior operator, either the SM or SRO, shall be on shift having had at least 6 months of hot operating experience.

6.2.3 (THIS SPECIFICATION NUMBER IS NOT USED)

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Control Room Commander in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the station.

6.3 (THIS SPECIFICATION NUMBER IS NOT USED)

6.4 (THIS SPECIFICATION NUMBER IS NOT USED)

6.5 REPORTABLE EVENT ACTION

The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the SORC and the results of this review shall be submitted to the Company Nuclear Review Board (CNRB) and the Site Vice President.

6.6 SAFETY LIMIT VIOLATION

The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Site Vice President and the CNRB shall be notified within 24 hours;

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC. This report shall describe: (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence;

c. The Safety Limit Violation Report shall be submitted to the Commission, the CNRB, and the Site Vice President within 14 days of the violation; and

d. Operation of the station shall not be resumed until authorized by the Commission.

6.7 PROCEDURES AND PROGRAMS

6.7.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Not used;

d. Not used;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation;

g. Quality Assurance Program for effluent and environmental monitoring;

h. Fire Protection Program implementation; and

i. Technical Specification Improvement Program implementation.

6.7.2 (THIS SPECIFICATION NUMBER IS NOT USED)

6.7.3 (THIS SPECIFICATION NUMBER IS NOT USED)

6.7.4 (THIS SPECIFICATION NUMBER IS NOT USED)

6.7.5 (THIS SPECIFICATION NUMBER IS NOT USED)

PROCEDURES AND PROGRAMS

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

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the RHR and containment spray, Safety Injection, chemical and volume control. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program that 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.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables,

2) Identification of the procedures used to measure the values of the critical variables,

3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

4) Procedures for the recording and management of data,

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

5) Procedures defining corrective actions for all off-control point chemistry conditions, and

6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program that will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1) Training of personnel, and

2) Procedures for monitoring.

e. (Not Used)

f. Accident Monitoring Instrumentation

A program which will ensure the capability to monitor plant variables and systems operating status during and following an accident. This program shall include those instruments provided to indicate system operating status and furnish information regarding the release of radioactive materials (Category 2 and 3 instrumentation as defined in Regulatory Guide 1.97, Revision 3)* and provide the following:

1) Preventive maintenance and periodic surveillance of instrumentation,

2) Preplanned operating procedures and backup instrumentation to be used if one or more monitoring instruments become inoperable, and

3) Administrative procedures for returning inoperable instruments to OPERABLE status as soon as practicable.

*Seabrook has taken exception to the categorization of instrumentation provided in Regulatory Guide 1.97, Revision 3. The Seabrook exceptions are provided in FSAR Table 7.5-1, which has been reviewed by the NRC staff in SER Supplement No. 5.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

g. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS, conforming to ten times the concentration values in Appendix B, Table 2, Column 2, to 10 CFR 20.1001-20.2402,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from the unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY shall be limited to the following:

   a) For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and
   b) For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to 1500 mrems/yr to any organ,

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) (Not Used), and

11) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

h. 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,

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

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.

i. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements and acceptance criteria, using methodologies described in applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or an absolute specific gravity within limits,
2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
3. A clear and bright appearance with proper color*;

b. Within 31 days following addition of the new fuel oil to the storage tank(s), verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil, and

c. Total particulate concentration of the stored fuel oil is < 10 mg/l when tested every 31 days using methodologies described in applicable ASTM Standards.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

* For fuel oil that has been dyed, the centrifuge method for quantifying water and sediment in distillate fuels specified in the applicable ASTM Standard is an acceptable method of performing this verification.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

j. Technical Specification (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license or

2. A change to the updated FSAR (UFSAR) or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 6.7.6j.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

6.8 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.8.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.8.1.1 A summary report of station startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the station.

The Startup Report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.8.1.2 Annual Reports covering the activities of the station as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions** (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions;

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

**This tabulation supplements the requirements of 10 CFR Part 20.2206.

6.8.1.2 (Continued)

b. The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (μCi/gm) and one other radio-iodine isotope concentration (μCi/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

c. Documentation of all challenges to the pressurizer power-operated relief valves (PORVs) and safety valves.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.8.1.3 The annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.8.1.4 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year of operation shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

MONTHLY OPERATING REPORTS

6.8.1.5 Routine reports of operating statistics and shutdown experience shall be submitted on monthly basis to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attn: Document Control Desk, with a copy to the NRC Regional Administrator, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.8.1.6.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

1. Cycle dependent Overpower ΔT and Overtemperature ΔT trip setpoint parameters and function modifiers for operation with skewed axial power profiles for Table 2.2-1 of Specification 2.2.1,

2. SHUTDOWN MARGIN limit for MODES 1,2, 3, and 4 for Specification 3.1.1.1,

3. SHUTDOWN MARGIN limit for MODE 5 for Specification 3.1.1.2,

4. Moderator Temperature Coefficient BOL and EOL limits, and 300 ppm surveillance limit for Specification 3.1.1.3,

5. Shutdown Rod Insertion limit for Specification 3.1.3.5,

6. Control Rod Bank Insertion limits for Specification 3.1.3.6,

7. AXIAL FLUX DIFFERENCE limits for Specification 3.2.1,

8. Heat Flux Hot Channel Factor, F_Q^{RTP} and K(Z) for Specification 3.2.2,

9. Nuclear Enthalpy Rise Hot Channel Factor, and $F^{RTP}_{\Delta H}$ for Specification 3.2.3.

The CORE OPERATING LIMITS REPORT shall be maintained available in the Control Room.

6.8.1.6.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-10266-P-A, Rev. 2 with Addenda (Proprietary) and WCAP-11524-A, Rev. 2 with Addenda (Nonproprietary), "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", March, 1987.

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

2. WCAP-10079-P-A, (Proprietary) and WCAP-10080-A (Nonproprietary), "NOTRUMP: A Nodal Transient Small Break and General Network Code", August 1985.

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

6.8.1.6.b (Continued)

3. YAEC-1363-A, "CASMO-3G Validation," April, 1988.

   YAEC-1659-A, "SIMULATE-3 Validation and Verification," September, 1988.

   WCAP-11596-P-A, (Proprietary), "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores", June, 1988.

   WCAP-10965-P-A, (Proprietary), "ANC: A Westinghouse Advanced Nodal Computer Code", September, 1986.

   Methodology for Specifications:
   3.1.1.1 - SHUTDOWN MARGIN for MODES 1,2, 3, and 4
   3.1.1.2 - SHUTDOWN MARGIN for MODE 5
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

4. Seabrook Station Updated Final Safety Analysis Report, Section 15.4.6, "Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant System".

   Methodology for Specifications:
   3.1.1.1 - SHUTDOWN MARGIN for MODES 1, 2, 3, and 4
   3.1.1.2 - SHUTDOWN MARGIN for MODE 5

5. YAEC-1241, "Thermal-Hydraulic Analysis of PWR Fuel Elements Using the CHIC-KIN Code", R. E. Helfrich, March, 1981.

   WCAP-14565-P, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis", April, 1997.

   Letter from T. H. Essig (NRC) to H. Sepp (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-14565-P, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis", January, 1999.

   Methodology for Specification:
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

6.8.1.6.b (Continued)

6. YAEC-1849P, "Thermal-Hydraulic Analysis Methodology Using VIPRE-01 For PWR Applications," October, 1992.

   WCAP-11397-P-A, (Proprietary), "Revised Thermal Design Procedure", April, 1989.

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

7. YAEC-1854P, "Core Thermal Limit Protection Function Setpoint Methodology For Seabrook Station," October, 1992

   WCAP-14551 -P, (Proprietary), "Westinghouse Setpoint Methodology for Protection Systems, Seabrook Nuclear Power Station Unit 1, 24 Month Fuel Cycle Evaluation", June, 1998.

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

8. YAEC-1856P, "System Transient Analysis Methodology Using RETRAN for PWR Applications," December, 1992.

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

6.8.1.6.b (Continued)

9. YAEC-1752, "STAR Methodology Application for PWRs, Control Rod Ejection, Main Steam Line Break," October, 1990.

   Methodology for Specification:
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

10. YAEC-1855PA, "Seabrook Station Unit 1 Fixed Incore Detector System Analysis," October, 1992.

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor
    3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

11. YAEC-1624P, "Maine Yankee RPS Setpoint Methodology Using Statistical Combination of Uncertainties - Volume 1 - Prevention of Fuel Centerline Melt," March, 1988.

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor
    3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

12. NYN-95048, Letter from T. C. Feigenbaum (NAESCo) to NRC, "License Amendment Request 95-05: Positive Moderator Temperature Coefficient", May 30, 1995.

    Methodology for Specification:
    3.1.1.3 - Moderator Temperature Coefficient

13. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report". April, 1995, (Westinghouse Proprietary).

    Methodology for Specification:
    3.2.2 - Heat Flux Hot Channel Factor

6.8.1.6.b (Continued)

14. WCAP-10216-P-A, Revision 1A (Proprietary), "Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification", February, 1994.

    WCAP-8385-P, (Proprietary), "Power Distribution Control and Load Following Procedures", September, 1974.

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor

15. WCAP-9272-P-A, (Proprietary), "Westinghouse Reload Safety Evaluation Methodology", July, 1985.

    Methodology for Specifications:
    3.1.1.1 - SHUTDOWN MARGIN for MODES 1,2,3, and 4
    3.1.1.2 - SHUTDOWN MARGIN for MODE 5
    3.1.1.3 - Moderator Temperature Coefficient
    3.1.3.5 - Shutdown Rod Insertion Limit
    3.1.3.6 - Control Rod Insertion Limits
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor
    3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

6.8.1.6.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT for each reload cycle, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and the Resident Inspector.

SPECIAL REPORTS

6.8.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attn: Document Control Desk, with a copy to the NRC Regional Administrator within the time period specified for each report.

6.9 (THIS SPECIFICATION NUMBER IS NOT USED)

6.10 RADIATION PROTECTION PROGRAM

6.10.1 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.

6.11 HIGH RADIATION AREA

6.11.1 Pursuant to paragraph 20.1601(c) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a) and (b), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less-than 1000 mR/h at 30 cm (12 in.) from the radiation source or from any surface that the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area; or

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the Radiation Work Permit.

6.11.2 In addition to the requirements of Specification 6.11.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 30 cm (12 in.) from the radiation source or from any surface that the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

HIGH RADIATION AREA

6.11.2 (Continued)

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.12 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by the Operational Quality Assurance Program (OQAP). 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 SORC and approval of the Station Director.

6.13 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by the Operational Quality Control Program (OQAP). 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 level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the SORC and the approval of the Station Director.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and each affected page shall indicate the revision number the change was implemented.

6.14 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.14.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Annual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the SORC. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the change is to be made;

*Licensees may choose to submit the information called for in this Specification as part of the FSAR update, pursuant to 10 CFR 50.71.

6.14.1 (Continued)

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the SORC.

b. Shall become effective upon review and acceptance by the SORC.

6.15 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, dated September 1995," as modified by the following exception:

a. NEI 94-01-1995, Section 9.2-3: The first ILRT performed after October 30, 1992 shall be performed no later than October 29, 2007.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.6 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.15% of primary containment air weight per day.

The provisions of SR 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

Overall air lock leakage rate acceptance criterion is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

Each containment 8-inch purge supply and exhaust isolation valve leakage rate acceptance criterion is $\leq 0.01\ L_a$ when tested at P_a.

2.0 Environmental Protection Issues

In the FES-OL (NUREG-0895) dated December, 1982, the staff considered the environmental impacts associated with the operation of Seabrook Station, Unit No. 1. No aquatic/water quality, terrestrial, or noise issues were identified.

Aquatic matters are addressed by the effluent limitations and monitoring requirements contained in NPDES Permit No. NH0020338 issued by the U. S. Environmental Protection Agency (Region I) as amended. The NRC will rely on the U.S.E.P.A and the NPDES Permit for regulation of matters involving water quality and aquatic biota.