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TO:	ORIG	CC	OTHER	
Mr. B.C. Rusche	3 signed	37		SENT NRC PDR XX SENT LOCAL PDR XX

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DESCRIPTION: Ltr notarized 11-6-75 re our 10-22-74 ltr....requesating for a revision of Tech Specs for Administrative Controls Section of Tech Specs & trans the following:

ENCLOSURES: Proposed Tech Specs Changes for H.B. Robinson Unit 2....

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ACKNOWLEDGED

PLANT NAME: H.B. Robinson Unit 2

FOR ACTION/INFORMATION

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Regulatory Docket File

November 6, 1975

FILE: NG-3514 (R)

SERIAL: NG-75-1578

50-261

RECEIVED NOV 10 1975 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

H. B. ROBINSON UNIT NO. 2
LICENSE NO. DPR-23
REVISION OF TECHNICAL SPECIFICATIONS FOR ADMINISTRATIVE CONTROLS

Dear Mr. Rusche:

In response to your letter of October 22, 1974, Carolina Power & Light Company submitted with our letter of December 17, 1974, a proposed revision to the "Administrative Controls" section of the H. B. Robinson Unit No. 2 Technical Specifications. We provided amendments to this proposed revision on February 3, 1975, and on September 8, 1975, to assure uniformity of plant and Company safety committee functions between the Robinson Plant and our Brunswick Plant, and to reflect a change in the plant organization. Recent discussions with your staff concerning several items in these submittals and the change in designation from abnormal occurrences to reportable occurrences have prompted us to consolidate the earlier submittals into one document and submit them for your approval. In addition to changing "abnormal" to "reportable" for the reporting requirements, we have expanded Section 6.9, "Reporting Requirements," to be consistent with Regulatory Guide 1.16 and the technical specifications requirements for reporting for our Brunswick Plant.

In addition, the Company requests that the Commission approve the restructuring of our offsite review of plant nuclear safety related activities. Currently, our offsite review is being handled by the Company Nuclear Safety Committee (CNSC). However, because of the increase in the number of operating nuclear plants on our system and the increasing scope of activities which require review, a determination was made that an alternative to committee review must be found. An independent nuclear safety review organization has been created and will assume the responsibilities and authority presently held by the CNSC. When this turnover has been completed, the CNSC will be dissolved.

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336 Fayetteville Street • P. O. Box 1551 • Raleigh, N. C. 27602

The independent nuclear safety review activity is patterned after the description given in Section 4, "Reviews and Audits," of American National Standards Institute (ANSI) N18.7 Rev. 1, Draft 5 dated February 12, 1975. The audit functions discussed in ANSI 18.7, Section 4, and those currently carried by the CNSC will be assigned to our Corporate Quality Assurance Audit (CQAA) Section. The independent nuclear safety review function discussed in ANSI 18.7, Section 4, and those currently carried by our CNSC will be assigned to a newly created Corporate Nuclear Safety (CNS) Section. Our proposed organization is the type actively encouraged for "nuclear oriented utilities" by paragraphs 3 and 4 under Section 4.1 of ANSI 18.7.

We believe that the movement from committee to organizational sections for review of nuclear safety related activities enhances our total nuclear safety program. Specific strengthening aspects of this change are:

1) By using personnel whose prime responsibility is the nuclear safety of the operating plants, we anticipate an enhancement in the depth and scope of review.

2) The increased sensitivity to needed actions by concentrating the responsibility for action on two individuals, Manager - Corporate Quality Assurance Audit and Manager - Corporate Nuclear Safety.

3) The independence of both the audit and review organizations from the line organization responsible for plant operation.

4) The increased responsiveness to plant conditions created by persons uniquely dedicated to the review and audit activities.

To assure that the organizational sections formed are suitable for fulfillment of their charged responsibilities, the CNSC will overlap with and monitor their activities for a period of time prior to dissolving itself as a formal body.

The consolidation of all outstanding proposed changes to the "Administrative Controls" section includes the proposed changes to delete the functions of the CNSC and incorporate the function of the CNS and CQAA Sections. The proposed revisions to the Technical Specifications are included as an attachment to this letter in the form of page changes to the Technical Specifications and supercedes all previous unapproved submittals on this topic. Vertical bars in the right-hand margin of the attached pages indicate changes which were not included in the earlier submittals referenced above.

Your prompt review and approval of this consolidated Technical Specification change is requested.

Yours very truly,

E. E. Utley
Vice President
Bulk Power Supply

RLM/nja

Attachment

bcc: Messrs. H. R. Banks, N. B. Bessac, P. W. Howe, J. A. Jones, W. B. Kincaid, L. I. Loflin, J. B. McGirt, D. B. Waters

Sworn to and subscribed before me this 6th day of November, 1975.

Margaret M. Cox
Notary Public

My Commission Expires: July 4, 1980

ENCLOSURE

Received w/Ltr Dated 11-6-75

H. B. ROBINSON UNIT 2

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

Section	Title	Page

1.6.2 Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating action.

1.6.3 Channel Calibration

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known value of the parameter which the channel measures. Calibration shall encompass the entire channel, including alarm or trip, and shall be deemed to include the channel functional test.

1.7 Containment Integrity

Containment integrity is defined to exist when:

a. All non-automatic containment isolation valves not required for normal operation are closed and blind flanges are properly installed where required.

b. The equipment door is properly closed and sealed.

c. At least one door in the personnel air lock is properly closed and sealed.

d. All automatic containment isolation trip valves are operable or are secured closed. Manual valves qualifying as automatic containment isolation valves are secured closed.

e. The uncontrolled containment leakage satisfies Specification 4.4.

1.8 Quadrant Power Tilt

The quadrant power tilt is defined as the ratio of maximum to average of the upper excore detector currents or the lower excore detector currents whichever is greater. If one excore is out of service, the three in-service units are used in computing the average.

6.0 Administrative Controls

6.1 Responsibility

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 Organization

Offsite

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

Facility Staff

6.2.2 The facility organization shall be as shown on Figure 6.2-2 and:

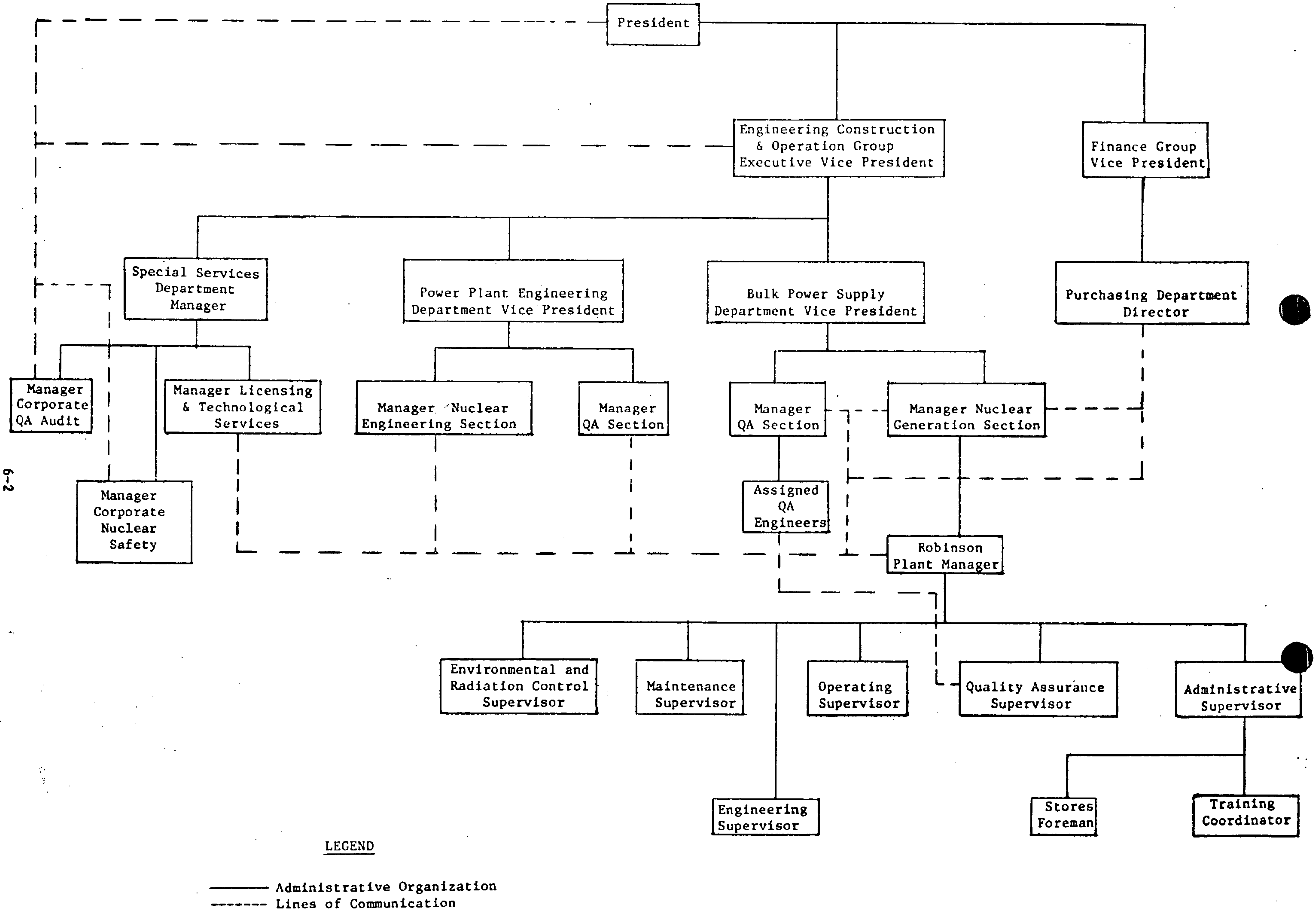
a. The shift complement shall consist of at least one shift foreman holding a Senior Reactor Operator's License, two control operators each holding a Reactor Operator's License, and one additional shift member.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. ALL CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.


President
Engineering Construction & Operation Group Executive Vice President
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Special Services Department Manager
Power Plant Engineering Department Vice President
Bulk Power Supply Department Vice President
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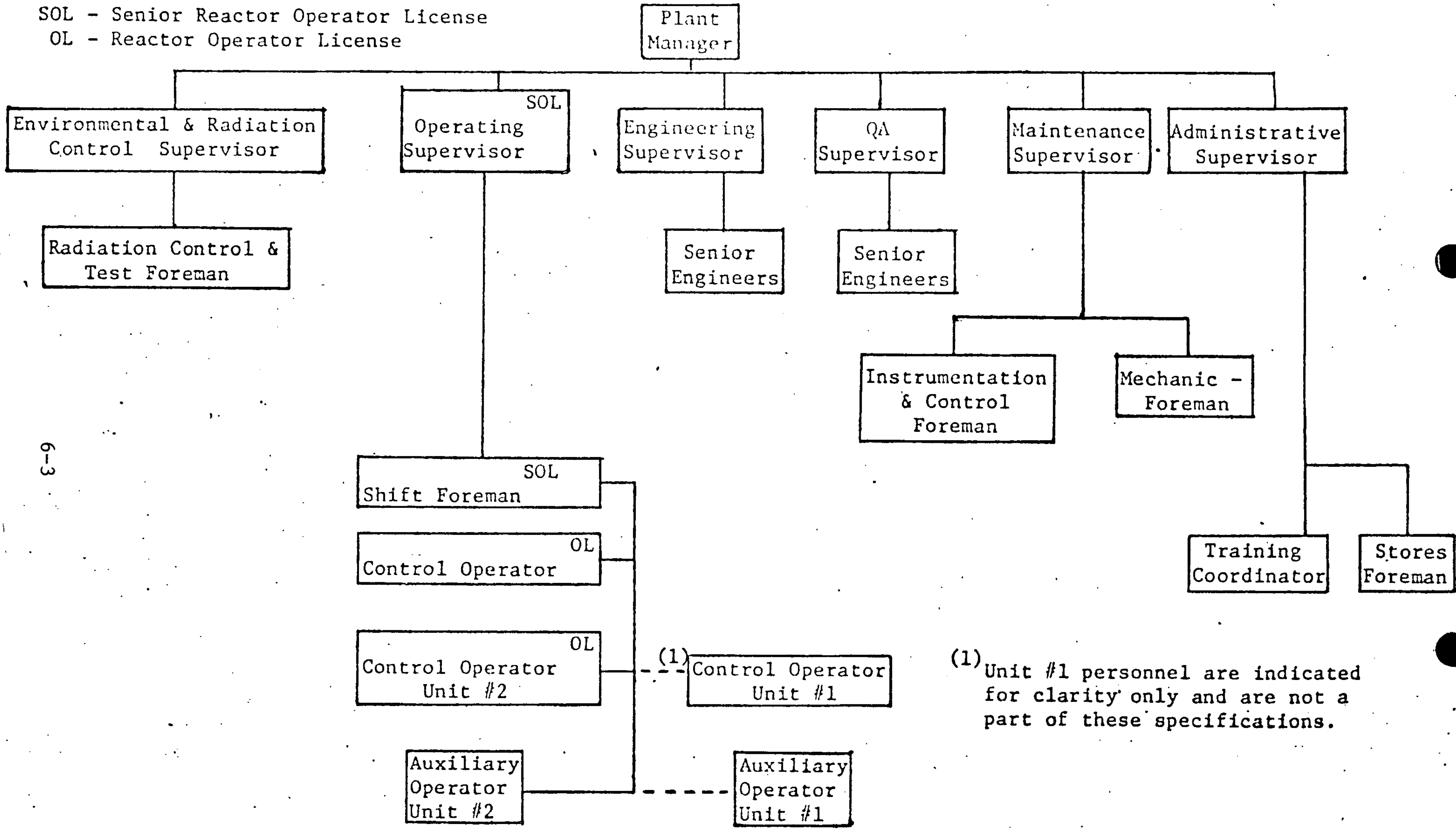
Manager Corporate QA Audit
Manager Licensing & Technological Services
Manager Corporate Nuclear Safety
Manager Nuclear Engineering Section
Manager QA Section
Manager QA Section
Manager Nuclear Generation Section
Assigned QA Engineers
Robinson Plant Manager
Environmental and Radiation Control Supervisor
Maintenance Supervisor
Operating Supervisor
Quality Assurance Supervisor
Administrative Supervisor
Engineering Supervisor
Stores Foreman
Training Coordinator
LEGEND
Administrative Organization
------- Lines of Communication

Figure 6.2-1

CONDUCT OF OPERATIONS CHART

SOL - Senior Reactor Operator License
OL - Reactor Operator License



(1) Unit #1 personnel are indicated for clarity only and are not a part of these specifications.

Figure 6.2-2

6.3 Facility Staff Qualifications

6.3.1 Each member of the facility staff shall meet or exceed ANSI N18.1-1971 with regard to the minimum qualifications for comparable positions.

6.4 Training

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Administrative Supervisor and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.5 Review and Audit

Organizational units for the review and audit of plant operations shall be constituted and have the responsibilities and authorities outlined below:

6.5.1 Plant Nuclear Safety Committee (PNSC)

6.5.1.1 Purpose

As an effective means for regular review, evaluation, and maintenance of plant operational safety, a Plant Nuclear Safety Committee (PNSC) has been established. The committee is chaired by the Plant Manager and composed of plant supervisory personnel.

6.5.1.2 Composition

The Plant Nuclear Safety Committee shall be composed of the following:

(a) Chairman: Plant Manager

(b) Vice Chairman:	Operating Supervisor
(c) Secretary:	Administrative Supervisor
(d)	Engineering Supervisor
(e)	Maintenance Supervisor
(f)	Environmental and Radiation Control Supervisor
(g)	Quality Assurance Supervisor

6.5.1.3 Alternates

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

6.5.1.4 Consultants

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

6.5.1.5 Meeting Frequency

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

6.5.1.6 Quorum

A quorum of the PNSC shall consist of the Chairman or Vice Chairman plus three members including alternates.

6.5.1.7 Responsibilities

a) Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

b) Review of all proposed test and experiments that affect nuclear safety.

c) Review of all proposed changes to the Technical Specifications.

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

e) Investigation of all violations of the Technical Specifications and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Manager of Nuclear Generation and to the Manager - Corporate Nuclear Safety.

f) Review of facility operations to detect potential safety hazards.

g) Performance of special reviews and investigations and reports thereon as requested by the Manager - Corporate Nuclear Safety.

h) Review of the Plant Security Plan and implementing procedures.

i) Review of the Emergency Plan and implementing procedures.

6.5.1.8 <u>Authority</u>

a) The Plant Nuclear Safety Committee shall be advisory.

b) The Plant Nuclear Safety Committee shall recommend to the Plant Manager approval or disapproval of proposals under 6.5.1.7a) through d) above.

In the event of disagreement between the recommendations of the Plant Nuclear Safety Committee and the actions contemplated by the Plant Manager, the course determined by the Plant Manager to be more conservative will be followed with immediate

notification to the Manager of Nuclear Generation and to the Manager - Corporate Nuclear Safety.

c) The Plant Nuclear Safety Committee shall make determinations as to whether or not proposals considered by the Committee involve unreviewed safety questions. This determination shall be subject to review by the Manager - Corporate Nuclear Safety as specified under 6.5.2.4.(a).

6.5.1.9 Records

Minutes shall be kept at the plant of all meetings of the Plant Nuclear Safety Committee and copies shall be sent to the Manager of Nuclear Generation and to the Manager - Corporate Nuclear Safety.

6.5.1.10 Procedures

Written administrative procedures for committee operation shall be prepared and maintained.

6.5.2 Independent Off-Site Safety Review Program

Activities occurring during plant operations shall be independently reviewed as specified in succeeding paragraphs.

6.5.2.1 Purpose

The purpose of the independent off-site safety review program is to review significant plant changes, tests, and procedures; verify that reportable occurrences are promptly investigated and corrected in a manner which reduces the probability of recurrence of such events; and detect trends which may not be apparent to a day-to-day observer.

6.5.2.2 Responsibility

The Manager - Corporate Nuclear Safety under the Department Manager - Special Services Department is charged with the overall responsibility for administering the independent off-site nuclear safety review program as follows:

a. Approves selection of the person or persons to conduct off-site safety reviews.

b. Has access to the plant operating records and operating personnel in performing the independent reviews.

c. Prepares and retains written records of reviews.

d. Assures independent safety review is conducted on all items required by Section 6.5.2.4.

e. Distributes reports and other records to appropriate managers.

6.5.2.3 Personnel

a. Personnel assigned responsibility for independent reviews shall be specified in technical disciplines, and shall collectively have the experience and competence required to review problems in the following areas:

    1. Nuclear power plant operations
    2. Nuclear engineering
    3. Chemistry and radiochemistry
    4. Metallurgy
    5. Instrumentation and control
    6. Radiological safety

7. Mechanical and electrical engineering
8. Administrative controls
9. Seismic and environmental
10. Quality assurance practices

b. The following minimum experience requirements shall be established for those persons involved in the independent off-site safety review program:

1. Manager - Bachelor of Science in engineering or related field and ten (10) years related experience including five (5) years involvement with operation and/or design of nuclear power plants.
2. Reviewers - Bachelor of Science in engineering or related field and five (5) years related experience including three (3) years involvement with operation and/or design of nuclear power plants.

c. An individual may possess competence in more than one specialty area. If sufficient expertise is not available with the Corporate Nuclear Safety Section, competent individuals from other CP&L organizations or outside consultants shall be utilized in performing independent off-site reviews and investigations.

d. Independent safety reviews shall be performed by personnel not directly involved with the activity or responsible for the activity.

6.5.2.4 Subjects Requiring Independent Review

The following subjects shall be reviewed by the Corporate Nuclear Safety Section:

a. Written safety evaluations, as required by 10CFR50.59(b), of changes in the facility, changes in procedures, and tests or experiments completed without prior NRC approval under the provisions of 10CFR50.59(a)(1) to the extent that such changes, tests, or experiments constituted changes in the facility, facility procedures, tests or experiments as described in the Safety Analysis Report. These reviews shall be conducted to verify that such changes, tests, or experiments did not involve a change in the technical specifications or an unreviewed safety question as defined in 10CFR50.59(a)(2).

b. Proposed changes in procedures, proposed changes in the facility, and proposed tests and experiments which have been determined to involve an unreviewed safety question as defined in 10CFR50.59(a)(2) prior to submitting the proposed changes to the NRC for consideration.

c. Proposed changes in the technical specifications or license amendments prior to submitting the proposed changes to the NRC for consideration.

d. Violations, deviations, and reportable occurrences such as:

1. Violations of applicable codes, regulations, orders, technical specifications, license requirements or internal procedures or instructions having safety significance;

2. Significant operating abnormalities or deviations from normal or expected performance of plant safety-related structures, systems, or components; and

3. Reportable occurrences pursuant to the requirements of Specification 6.11.2.a.

Review of events covered under this paragraph shall include the results of any investigations made and the recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event.

e. Reports by and meeting minutes of the Plant Nuclear Safety Committee.

f. Other matters affecting the nuclear safety of the nuclear power plant which the Manager - Corporate Nuclear Safety deems appropriate for consideration, or which are referred to the independent reviewers by the onsite operating organization or by other functional organizational units within Carolina Power & Light Company.

6.5.2.5 Follow-up Action

Results of Corporate Nuclear Safety reviews, including recommendations and conclusions will be documented. Recommendations and conclusions will be submitted to the Manager - Nuclear Generation with copies to Company President; Group Executive - Engineering, Construction & Operation Group; Department Head - Bulk Power Supply; Department Head - Special Services; Plant Manager and others as appropriate. Copies of the documented review will be retained in the Corporate Nuclear Safety Section files.

6.5.2.6 The Corporate Nuclear Safety review program shall be conducted in accordance with written, approved procedures.

6.5.3 Independent Off-Site Quality Assurance Audit Program

6.5.3.1 Purpose

A comprehensive system of planned and documented audits shall be carried out to verify compliance with all aspects of the administrative controls and quality assurance program. Audits of selected aspects of operational phase activities shall be performed with a frequency commensurate with their safety significance and in such a manner as to assure that an audit of all safety-related functions is completed within a period of two years.

Audits should include verification of compliance and effectiveness of implementation of internal rules; procedures (for example: operating, design, procurement, maintenance, modification, refueling, surveillance, test, security and radiation control procedures and the emergency plan); regulations and license provisions; off-site nuclear safety reviews; programs for training, retraining, and qualification of the operating

staff; corrective actions taken following reportable occurrences and observation of operating, refueling, maintenance and modification activities, including associated record keeping.

6.5.3.2 Responsibility

The Manager - Corporate Quality Assurance Audit is charged with the overall responsibility for the corporate quality assurance audit program as follows:

a. Selects auditors

b. Has access to records and personnel necessary in performing the audits.

6.5.3.3 Personnel

a. Audit personnel will be independent of the area audited. Selection for auditing assignments is based on experience or training which establishes that their qualifications are commensurate with the complexity or special nature of the activities to be audited. In selecting auditing personnel, consideration will be given to special abilities, specialized technical training, prior pertinent experience, personal characteristics, and education.

b. Qualified outside consultants or other individuals within the EC&O Group will be used to augment the audit teams when necessary.

6.5.3.4 Reports

Results of audit are approved by the Manager - Corporate Quality Assurance Audit and transmitted directly to the Company President and the Group Executive - Engineering, Construction

& Operation Group, as well as, to the Department Head - Bulk Power Supply, Department Head - Special Services, and others, as appropriate.

6.5.3.5 The corporate quality assurance audit program shall be conducted in accordance with written, approved procedures.

6.6 Reportable Occurrence Action

6.6.1 The following actions shall be taken in the event of a reportable occurrence:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each Reportable Occurrence Report submitted to the Commission shall be reviewed by the PNSC and submitted to the Manager of Corporate Nuclear Safety and the Manager of Nuclear Generation.

6.7 Safety Limit Violation

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

a. The provisions of 10 CFR 50.36 (c) (1) (i) shall be complied with immediately.

b. The Safety Limit violation shall be reported to the Commission, the Manager of Nuclear Generation and the Manager of Corporate Nuclear Safety within 24 hours.

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

d. The Safety Limit Violation Report shall be submitted to the Commission, the Manager of Corporate Nuclear Safety and the Manager, Nuclear Generation within 14 days of the violation.

6.8 Procedures

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix "A" of USNRC Regulatory Guide 1.33 dated 11/3/72 except as provided in 6.8.2 and 6.8.3 below.

6.8.2 Proposed operating procedures, overall plant operating procedures, system descriptions, emergency procedures, fuel handling procedures, periodic test procedures, procedures for equipment maintenance which may affect nuclear safety, annunciator procedures and any other procedures determined by the Plant Manager to affect nuclear safety, shall be reviewed by the PNSC and approved by the Plant Manager. Prior to implementation, proposed changes to these procedures must also be reviewed and approved in this manner.

6.8.3 Temporary changes to procedures of 6.8.2 above 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 Reactor Operator's License.

c. The change is documented, reviewed by the PNSC and approved by the Plant Manager within three weeks of implementation.

6.9 Reporting Requirements

Information to be reported to the Commission, in addition to the reports required by Title 10, Code of Federal Regulations,

shall be as indicated in the following sections. Reports shall be addressed to the Director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.

6.9.1 Routine Reports

a. Startup Report. A summary report of plant startup and power escalation 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 plant. The report shall address each of the tests identified in the FSAR 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.

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 power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

b. <u>Annual Operating Report.</u>[1,2] Routine operating reports covering the operation of the unit during 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.

The primary purpose of annual operating reports is to permit annual evaluation by the NRC staff of operating and maintenance experience throughout the nuclear power industry. The annual operating reports made by licensees shall provide a comprehensive summary of the operating experience gained during the year, even though some repetition of previously reported information may be involved. References in the annual operating report to previously submitted reports shall be clear.

Each annual operating report shall include:

(1) A narrative summary of operating experience during the report period relating to safe operation of the facility, including safety-related maintenance not covered in 6.9.1.(2)(e) below.

(2) For each outage or forced reduction in power[3] of over twenty percent of design power level where the reduction extends for greater than four hours:

(a) the proximate cause and the system and major component involved (if the outage or forced reduction in power involved equipment malfunction);

1/ 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.

2/ Much of the information in the Annual Report was previously submitted in a Semiannual Report.

3/ The term "forced reduction in power" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the load on the unit be reduced for corrective action immediately or up to and including the very next weekend. Note that routine preventive maintenance, surveillance and calibration activities requiring power reductions are not covered by this section.

(b) a brief discussion of (or reference to reports of) any reportable occurrences pertaining to the outage or power reduction;

(c) corrective action taken to reduce the probability of recurrence, if appropriate;

(d) operating time lost as a result of the outage or power reduction (for scheduled or forced outages,[4/] use the generator off-line hours; for forced reductions in power, use the approximate duration of operation at reduced power);

(e) a description of major safety-related corrective maintenance performed during the outage or power reduction, including the system and component involved and identification of the critical path activity dictating the length of the outage or power reduction; and

(f) a report of any single release of radioactivity or single radiation exposure specifically associated with the outage which accounts for more than 10% of the allowable annual values.

(3) A tabulation of man rem for (Supplementing the requirements of § 20.407 of 10 CFR Part 20) the tabulated number of personnel receiving exposures greater than 100 mrem in the reporting period according to duty function, e.g., routine plant surveillance and inspection (regular duty), routine plant maintenance, special plant maintenance (describe maintenance), routine fueling operation, special refueling operation (describe operation), and other job-related exposures. Estimates of the dose

4/ The term "forced outage" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the unit be removed from service for corrective action immediately or up to and including the very next weekend.

assignment to various duty functions shall be based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the total dose need not be individually accounted for, however, in the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific work functions. See Appendix A to Regulatory Guide 1.16* for the required format for providing this information.

(4) Findings from irradiated fuel examinations, including results of eddy current tests, ultrasonic tests, or visual examinations completed during the report period.

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis. The report formats set forth in Appendices B, C, and D to Regulatory Guide 1.16* shall be completed in accordance with the instructions provided. The completed forms should be submitted by the tenth of the month following the calendar month covered by the report to the Director, Office of Management Information and Program Control, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the appropriate NRC Regional Office.

6.9.2 Reportable Occurrences

Guidance concerning reportable occurrences that shall be reported in different time frames are provided below:

a. Prompt Notification With Written Followup. The types of events listed below shall be reported as expeditiously as possible but within 24 hours by telephone and confirmed by

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* Regulatory Guide 1.16, "Reporting of Operating Information Appendix A Technical Specifications," Revision 4.

telegraph, mailgram, or facsimile transmission to the Director of the appropriate Regional Office of Inspection and Enforcement or his designate no later than the first working day following the event, with a written followup within two weeks. A copy of the confirmation and the written followup report shall also be sent to the Director, Office of Management Information and Program Control, U. S. Nuclear Regulatory Commission. The written followup report shall include, as a minimum, a completed copy of the licensee event report form (see Appendix E to Regulatory Guide 1.16*) used for entering data into the NRC's computer-based file of information concerning licensee events. (Instructions for completing these licensee event report forms[5/] are issued individually to each licensee.) Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Failure of the reactor protection system, or other systems subject to limiting safety system settings, to initiate the required protective function by the time a monitored parameter reaches the value specified as the limiting safety system setting in the Technical Specifications, or failure to complete the required protective function.

Note: Instrument drift discovered as a result of testing need not be reported under this item (but see 6.9.2.a(5), 6.9.2.a(6), and 6.9.2.b(1) below).

(2) Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the Technical Specifications.

5/ Instruction Manual, Licensee Event Report File, Office of Management Information and Program Control, USNRC, Washington, D. C. 20555.

* Regulatory Guide 1.16, "Reporting of Operating Information Appendix A Technical Specifications," Revision 4.

Note: If specified action is taken when a system is found to be operating between the most conservative and least conservative aspects of a limiting condition for operation listed in the Technical Specifications, the limiting condition for operation is not considered to have been violated and no report need be submitted under this section (but see 6.9.2.b(2) below).

(3) Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary or primary containment.

Note: Leakage of valve packing or gaskets within the limits for identified leakage set forth in Technical Specifications need not be reported under this section.

(4) Reactivity anomalies involving disagreement with predicted value of reactivity balance under steady state conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a shutdown margin less than specified in the Technical Specifications; short-term reactivity increases that correspond to a reactor startup rate greater than 5 dpm, or if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or any unplanned criticality.

(5) Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

(6) Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

Note: For 6.9.2.a(5) and 6.9.2.a(6) reduced redundancy that does not result in loss of system function need not be reported under this section (but see 6.9.2.b(2) and 6.9.2.b(3) below).

(7) Conditions arising from natural or man-made events that, as a direct result of the event, require plant shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

(8) Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

(9) Performance of structures, systems or components that require remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or Technical Specifications bases or discovery during plant life of conditions not specifically considered in the safety analysis report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

Note: This item is intended to provide for reporting of potentially generic problems.

b. Thirty-day Written Reports. The reportable occurrences discussed below shall be the subject of written reports to the Director of the appropriate NRC Regional Office within thirty days of occurrence of the event. A copy of the written report should also be sent to the Director, Office of Management Information and Program Control. The written report shall include, as a minimum, a completed copy of the licensee event report form, (see Appendix E to Regulatory Guide 1.16*) used for entering data into the NRC's computer-based file of information concerning

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* Regulatory Guide 1.16, "Reporting of Operating Information Appendix A Technical Specifications," Revision 4.

licensee events. (Instructions for completing these licensee event report forms[5/] are issued individually to each licensee.) Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems (but see 6.9.2.a(1) and 6.9.2.a(2) above).

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation (but see 6.9.2.a(2) above).

Note: Routine surveillance testing, instrument calibration or preventive maintenance which require system configurations as described in 6.9.2.b(1) and 6.9.2.b(2) above need not be reported except where test results themselves reveal a degraded mode as described above.

(3) Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems (but see 6.9.2.a(6) above).

(4) Abnormal degradation of systems other than those specified in 6.9.2.a(3) above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in Technical Specifications need not be reported under this item.

6.9.3 Special Reports

Special reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

Area	Reference	Submittal Date
a. Containment Leak Rate Testing	4.4	Upon completion of each test
b. Initial Containment Structural Test	4.4	Within three months following completion of test
c. Fuel Inspection	2.1	Upon completion of the inspection at second and third refueling outages
d. Inservice Inspection Evaluation	4.2	After five years of operation
e. Containment Sample Tendon Survillance	4.4	Upon completion of the inspection at 5 and 25 years of operation
f. Post-operational Containment Structural Test	4.4	Upon completion of the test at 3 and 20 years of operation.

6.10 Record Rentention

6.10.1 The following records shall be retained for at least five years:

a. Records of facility operation covering time interval at each power level.

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

c. Abnormal Occurrence and Reportable Occurrence Reports.

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

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak test and results.

i. Records of annual physical inventory of all source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Record and drawing changes reflecting facility 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 facility components designed for a limited number of transients or cycles.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

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

k. Records of meetings of the PNSC and the CNSC.

6.11 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.

6.12 Respiratory Protection Program

Allowance

6.12.1 Pursuant to 10 CFR 20.103(c)(1) and (3), allowance may be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this facility in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table 1, Column 1, of 10 CFR 20, subject to the following conditions and limitations:

a. The limits provided in Section 20.103(a) and (b) shall not be exceeded.

b. If the radioactive material is of such form that intake through the skin or other additional route is likely, individual exposures to radioactive material shall be controlled so that the radioactive content of any critical organ from all routes of intake averaged over seven consecutive days does not exceed that which would result from inhaling such radioactive material for 40 hours at the pertinent concentration values provided in Appendix B, Table I, Column I, of 10 CFR 20.

c. For radioactive materials designated "Sub" in the "Isotope" column of Appendix B, Table I, Column 1 of 10 CFR 20, the concentration value specified shall be based upon exposure to the material as an external radiation source. Individual exposures to these materials shall be accounted for as part of the limitation on individual dose in §20.101. These materials shall be subject to applicable process and other engineering controls.

Protection Program

6.12.2 In all operations in which adequate limitation of the inhalation of radioactive material by the use of process or other engineering controls is impracticable, the licensee may permit an individual in a restricted area to use respiratory protective equipment to limit the inhalation of airborne radioactive material, provided:

a. The limits specified in 6.12.1 above, are not exceeded.

b. Respiratory protective equipment is selected and used so that the peak concentrations of airborne radioactive material inhaled by an individual wearing the equipment do not exceed the pertinent concentration values specified in Appendix B, Table I, Column 1, of 10 CFR 20. For the purposes of this subparagraph, the concentration of radioactive material that is inhaled when respirators are worn may be determined by dividing the ambient airborne concentration by the protection factor specified in

Table 6.12-1 for the respirator protective equipment worn. If the intake of radioactivity is later determined by other measurements to have been different than that initially estimated, the later quantity shall be used in evaluating the exposures.

c. The Licensee advises each respirator user that he may leave the area at any time for relief from respirator use in case of equipment malfunction, physical or psychological discomfort, or any other condition that might cause reduction in the protection afforded the wearer.

d. The licensee maintains a respiratory protective program adequate to assure that the requirements above are met. Such a program shall include:

   1. Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposures, and to permit proper selection of respiratory protective equipment.

   2. Written procedures to assure proper selection, supervision, and training of personnel using such protective equipment.

   3. Written procedures to assure the adequate fitting of respirators; and the testing of respiratory protective equipment for operability immediately prior to use.

   4. Written procedures for maintenance to assure full effectiveness of respiratory protective equipment, including issuance, cleaning and decontamination, inspection, repair, and storage.

   5. Written operational and administrative procedures for proper use of respiratory protective equipment including provisions

for planned limitations on working times as necessitated by operational conditions.

6. Bioassays and/or whole body counts of individuals (and other surveys, as appropriate) to evaluate individual exposures and to assess protection actually provided.

e. The licensee shall use equipment approved by the U. S. Bureau of Mines under its appropriate Approval Schedules as set forth in Table 6.12-1. Equipment not approved under U. S. Bureau of Mines Approval Schedules shall be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U. S. Bureau of Mines approved equipment of the same type, as specified in Table 6.12-1.

f. Unless otherwise authorized by the Commission, the licensee shall not assign protection factors in excess of those specified in Table 6.12-1 in selecting and using respiratory protective equipment.

Revocation

6.12.3 The specifications of Section 6.12 shall be revoked in their entirety upon adoption of the proposed change to 10 CFR 20, Section 20.103, which would make such provisions unnecessary.

6.13 High Radiation Area

6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and

entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continously indicates the radiation dose rate in the area.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.13.1(a) above, and in addition locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Foreman on duty.

TABLE 6.12-1
PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION	MODES[1]	PROTECTION FACTORS[2] PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE[3]	GUIDES TO SELECTION OF EQUIPMENT BUREAU OF MINES APPROVAL SCHEDULES* FOR EQUIPMENT CAPABLE OF PROVIDING AT LEAST EQUIVALENT PROTECTION FACTORS *or schedule superseding for equipment of type listed
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask[4,7]	NP	5	21B 30 CFR § 14.4(b) (4)
Facepiece, full[7]	NP	100	21B 30 CFR § 14.4(b) (5); 14F 30 CFR 13
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline respirator			
Facepiece, half-mask	CF	100	19B 30 CFR § 12.2(c) (2) Type C(i)
Facepiece, full	CF	1,000	19B 30 CFR § 12.2(c) (2) Type C(i)
Facepiece, full[7]	D	100	19B 30 CFR § 12.2(c) (2) Type C(ii)
Facepiece, full	PD	1,000	19B 30 CFR § 12.2(c) (2) Type C(iii)
Hood	CF	[5]	[6]
Suit	CF	[5]	[6]
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full[7]	D	100	13E 30 CFR § 11.4(b) (2) (i)
Facepiece, full	PD	1,000	13E 30 CFR § 11.4(b) (2) (ii)
Facepiece, full	R	100	13E 30 CFR § 11.4(b) (1)
III. COMBINATION RESPIRATOR			
Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above.	19B CFR § 12.2(e) or applicable schedules as listed above

1,2,3,4,5,6,7 (These notes are on the following pages)

TABLE 6.12-1 (Continued)

[1]See the following symbols:

CF: continuous flow

D: demand

NP: negative pressure (i.e., negative phase during inhalation)

PD: pressure demand (i.e., always positive pressure)

R: recirculating (closed circuit)

[2](a) For purposes of this specification the protection factor is a measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne radioactive material outside the respiratory protective equipment to that inside the equipment (usually inside the facepiece) under conditions of use. It is applied to the ambient airborne concentration to estimate the concentration inhaled by the wearer according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Ambient Airborne Concentration}}{\text{Protection Factor}}$$

(b) The protection factors apply:

(i) only for trained individuals wearing properly fitted respirators used and maintained under supervision in a well-planned respiratory protective program.

(ii) for air-purifying respirators only when high efficiency [above 99.9% removal efficiency by U. S. Bureau of Mines type dioctyl phthalate (DOP) test] particulate filters and/or sorbents appropriate to the hazard are used in atmospheres not deficient in oxygen.

(iii) for atmosphere-supplying respirators only when supplied with adequate respirable air.

[3] Excluding radioactive contaminants that present an absorption or submersion hazard. For tritium oxide approximately half of the intake occurs by absorption through the skin so that an overall protection factor of not more than approximately 2 is appropriate when atmosphere-supplying respirators are used to protect against tritium oxide. Air-purifying respirators are not recommended for use against tritium oxide. See also footnote [5] below, concerning supplied-air suits and hoods.

[4] Under chin type only. Not recommended for use where it might be possible for the ambient airborne concentration to reach instantaneous values greater than 50 times the pertinent values in Appendix B, Table I, Column 1 of 10 CFR Part 20.

[5] Appropriate protection factors must be determined taking account of the design of the suit or hood and its permeability to the contaminant under conditions of use. No protection factor greater than 1,000 shall be used except as authorized by the Commission.

[6] No approval schedules currently available for this equipment. Equipment must be evaluated by testing or on basis of available test information.

[7] Only for shaven faces.

NOTE 1: Protection factors for respirators, as may be approved by the U. S. Bureau of Mines according to approval schedules for respirators to protect against airborne radionuclides, may be used to the extent that they do not exceed the protection factors listed in this Table. The protection factors in this Table may not be appropriate to circumstances where chemical or other respiratory hazards exist in addition to radioactive hazards. The selection and use of respirators for such circumstances should take into account approvals of the U. S. Bureau of Mines in accordance with its applicable schedules.

NOTE 2: Radioactive contaminants for which the concentration values in Appendix B, Table 1 of this part are based on internal dose due to inhalation may, in addition, present external exposure hazards at higher concentrations. Under such circumstances, limitations on occupancy may have to be governed by external dose limits.