UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DUKE POWER COMPANY

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

SALUDA RIVER ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 80
License No. NPF-35

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-35 filed by the Duke Power Company acting for itself, North Carolina Electric Membership Corporation and Saluda River Electric Cooperative, Inc. (licensees) dated October 28, 1987, and supplemented October 30 and November 10, 1989, July 13 and September 25, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-35 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 80 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DB Matthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification Changes

Date of Issuance: November 29, 1990

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DUKE POWER COMPANY

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 74
License No. NPF-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-52 filed by the Duke Power Company acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees) dated October 28, 1987, and supplemented October 30 and November 10, 1989, July 13 and September 25, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-52 is hereby amended to read as follows:

    Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 74 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification Changes

Date of Issuance: November 29, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 80

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

TO LICENSE AMENDMENT NO. 74

FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Append'x "A" Technical Specifications with the enclosed pages. The revised pages ar: identified by Amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
VI	VI
IX	IX
X	X
XIII	XIII
XIV	XIV
3/4 3-18	3/4 3-18
3/4 3-19	3/4 3-19
3/4 7-39	3/4 7-39
3/4 7-40*	3/4 7-40
3/4 8-7	3/4 8-7
5-1*	5-1
5-2	5-2
5-3*	5-3
5-4	5-4
5-5	5-5
6-6a	6-6a
6-7	6-7
6-8	6-8
6-11	6-11
6-12	6-12
6-15	6-15
6-19	6-19
6-19a	6-19a
6-20	6-20

* Overleaf Pages

3/4.4 REACTOR COOLANT SYSTEM

BASES

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
e. Steam Line Pressure - Negative Rate-High	3/steam line	2/steam line in any steam line	2/steam line	3##	15
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	27
b. Steam Generator Water Level-High-High (P-14)	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2	19
c. T_{avg}-Low (P-4 Interlock)	4	2	3	1, 2	19
d. Doghouse Water Level-High	2/doghouse	1/doghouse in any doghouse	2/doghouse	1, 2	27
e. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
6. Turbine Trip					
a. Manual Initiation	1	1	1	1,2	25
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1,2	27

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Turbine Trip (Continued)					
c. Steam Generator Water Level-High-High (P-14)	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1,2	19
d. Trip of All Main Feedwater Pumps	2/pump	1/pump	1/pump	1,2#	25
e. Reactor Trip (P-4)	2	1	2	1,2,3	22
f. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
7. Containment Pressure Control System					
a. Start Permissive	4/train	2/train	3/train	1, 2, 3, 4	19
b. Termination	4/train	2/train	3/train	1, 2, 3, 4	19
8. Auxiliary Feedwater					
a. Manual Initiation	1/train	1/train	1/train	1, 2, 3	26
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21

SURVEILLANCE REQUIREMENTS

4.7.12 The groundwater level shall be determined at the following frequencies by monitoring the water level and by verifying the absence of alarm in the six groundwater monitor wells as shown in FSAR Figure 2.4.13-14 installed around the perimeter of the Reactor and Auxiliary Buildings:

a. At least once per 7 days when the groundwater level is at or below the top of the adjacent floor slab, and

b. At least once per 24 hours when the groundwater level is above the top of the adjacent floor slab.

3/4.7.13 STANDBY SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.13 The Standby Shutdown System (SSS) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION: (Units 1 and 2)

a. With the Standby Shutdown System inoperable, restore the inoperable equipment to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

b. With the total leakage from UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE and reactor coolant pump seal leakage greater than 26 gpm, declare the Standby Makeup Pump inoperable and take ACTION a., above.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.13.1 The Standby Shutdown System diesel generator shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

   1) The fuel level in the fuel storage tank is greater than or equal to 67 inches, and

   2) The diesel starts from ambient conditions and operates for at least 30 minutes at greater than or equal to 700 kW.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-1975, is within the acceptable limits specified in Table 1 of ASTM-D975-1977 when checked for viscosity and water and sediment; and

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.13.2 The Standby Shutdown System diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1) The electrolyte level of each battery is above the plates; and

   2) The overall battery voltage is greater than or equal to 24 volts.

11) Verifying that the fuel transfer valve transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines;

12) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within the tolerances given in Table 4.8-2;

13) Verifying that the voltage and diesel speed tolerances for the accelerated sequencer permissives are 92.5 ± 1% and 98 + 1%, respectively, with a time delay of 2 ± 0.2 s;

14) Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

   a) Turning gear engaged, or

   b) Maintenance mode; and

15) Operating at greater than or equal to 5600 KW but less than or equal to 5750 KW for one hour or until operating temperature has stabilized. Within 5 minutes after completing this test, perform Specification 4.8.1.1.2g.6)b).

h. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 441 rpm in less than or equal to 11 seconds; and

i. At least once per 10 years by:

1) Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or its equivalent, and

2) Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110% of the system design pressure.

3) Performing tank wall thickness measurements. The resulting data shall be evaluated and any abnormal degradation will be justified or corrected. Any abnormal degradation will be documented in a report to the Commission.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests of any diesel generator is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The Exclusion Area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure 5.1-2.

MAPS DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figures 5.1-3 and 5.1-4.

The definition of UNRESTRICTED AREA used in implementing these Technical Specifications has been expanded over that in 10 CFR 20.3(a)(17). The UNRESTRICTED AREA boundary may coincide with the Exclusion (fenced) Area boundary, as defined in 10 CFR 100.3(a), but the UNRESTRICTED AREA does not include areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment structure is comprised of a steel containment vessel surrounded by a concrete containment having the following design features:

a. Containment Vessel

1) Nominal inside diameter = 115 feet.
2) Nominal inside height = 171 feet.
3) Nominal thickness of vessel walls = 0.75 inch.
4) Nominal thickness of vessel dome = 0.6875 inch.
5) Nominal thickness of vessel bottom = 0.25 inch.
6) Net free volume = 1.2×10^6 cubic feet.

b. Reactor Building

1) Nominal Annular space = 6 feet.
2) Annulus nominal volume = 484,090 cubic feet.
3) Nominal outside height (top of foundation base to top of dome) = 177 feet.
4) Nominal inside diameter = 127 feet.
5) Minimum cylinder wall thickness = 3 feet.
6) Minimum dome thickness = 2.25 feet.
7) Dome inside radius = 87 feet.

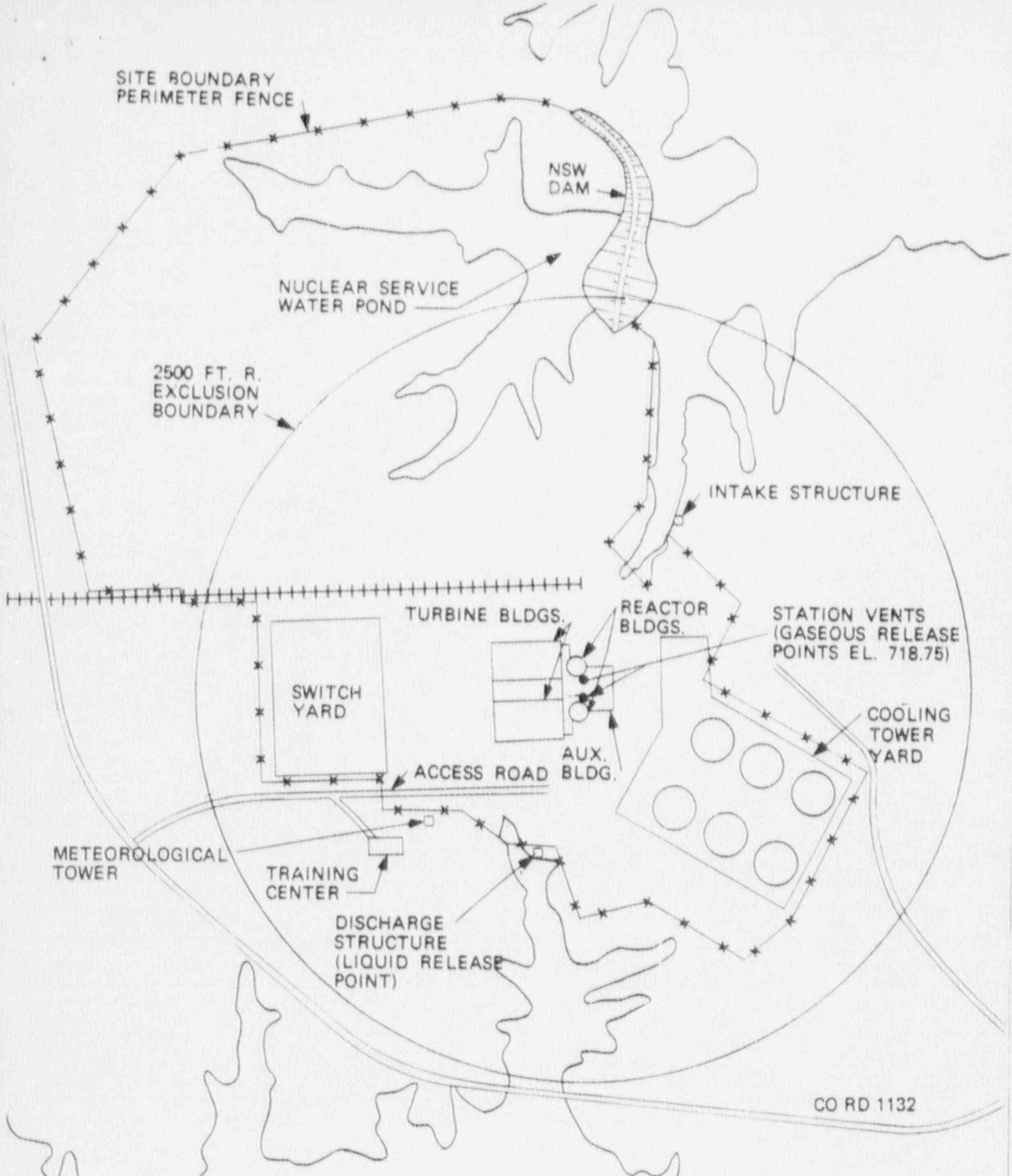

FIGURE 5.1-1
EXCLUSION AREA

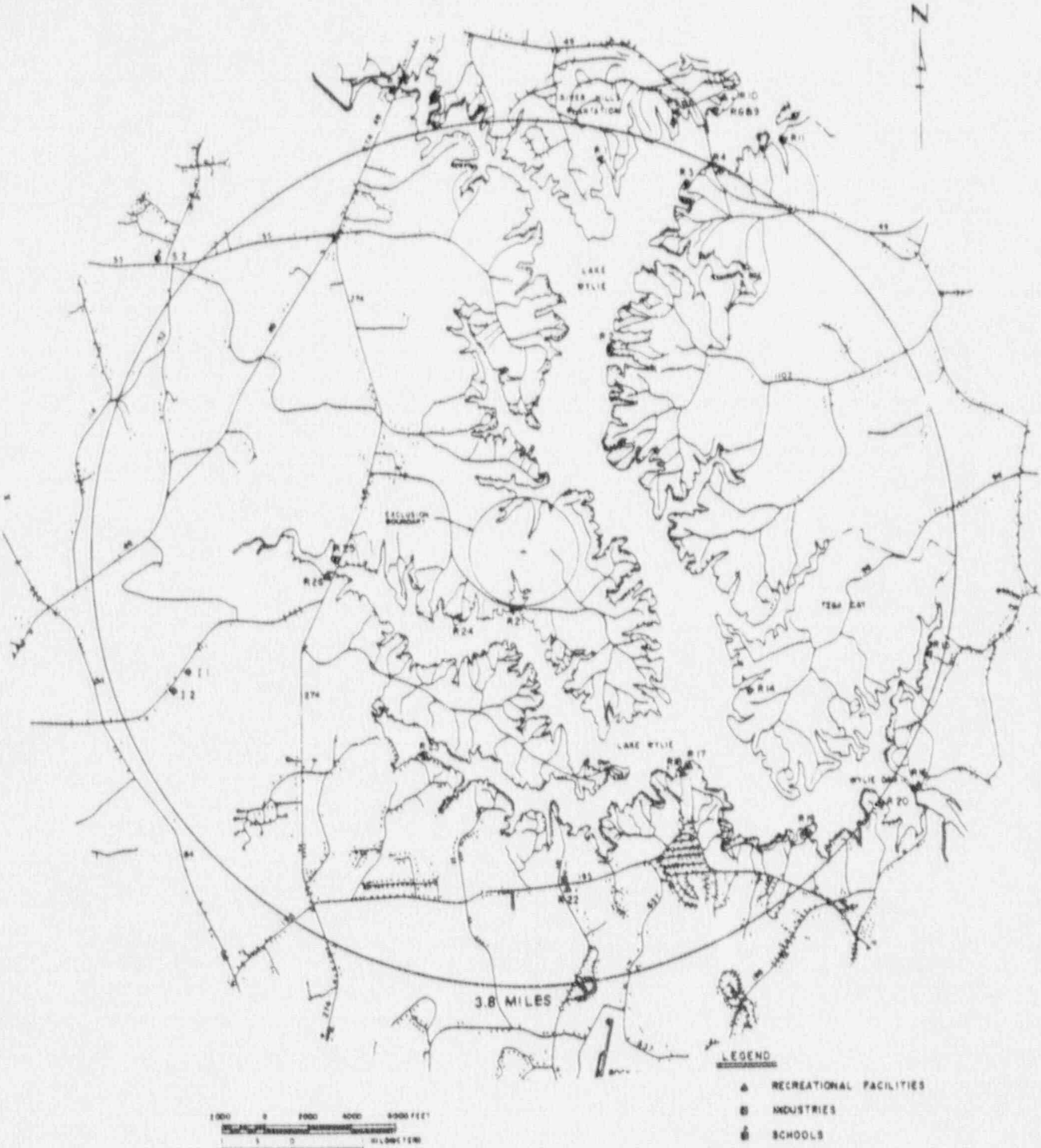

FIGURE 5.1-2
LOW POPULATION ZONE

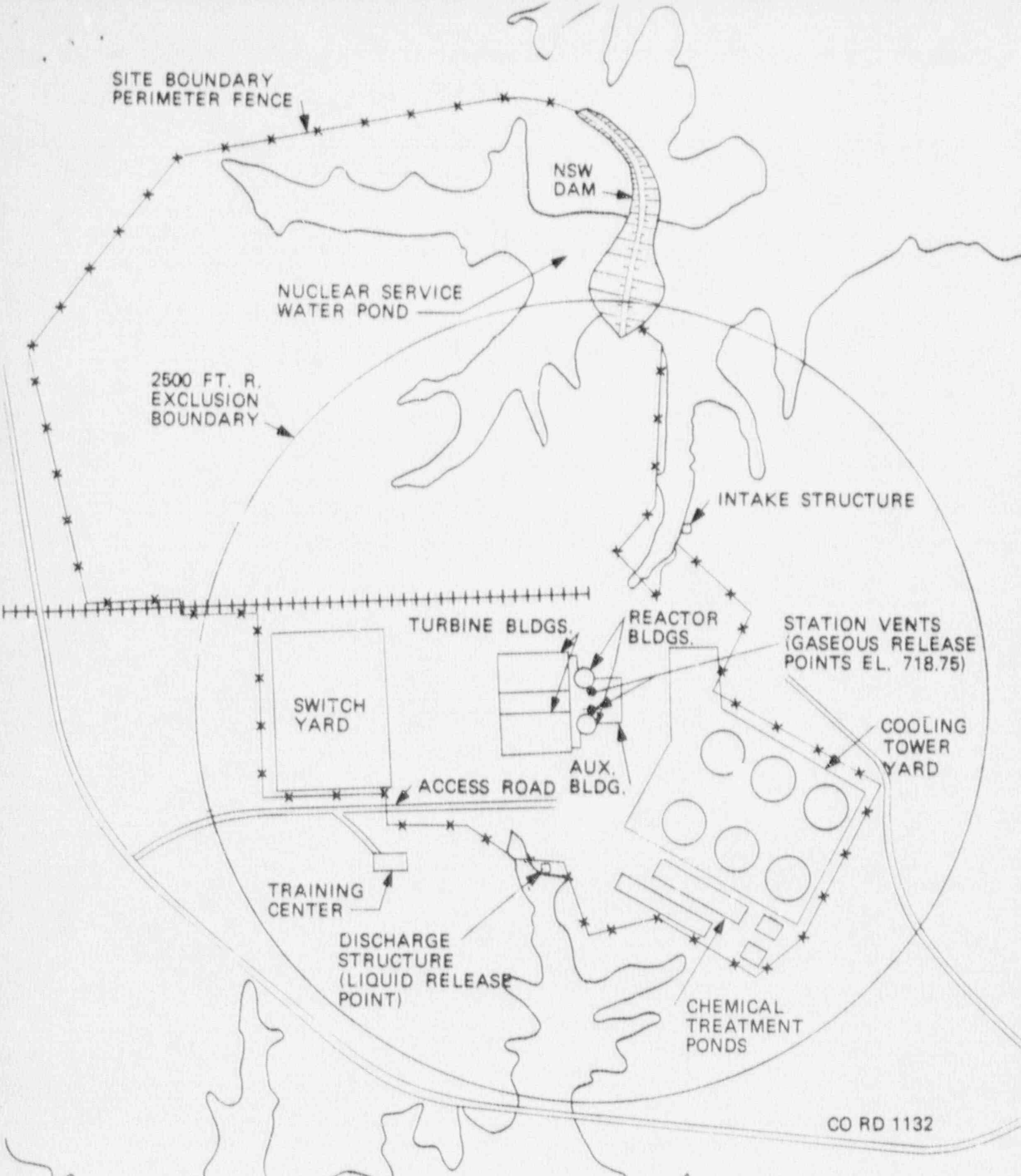

FIGURE 5.1-3

UNRESTRICTED AREA AND SITE BOUNDARY FOR RADIOACTIVE LIQUID EFFLUENTS

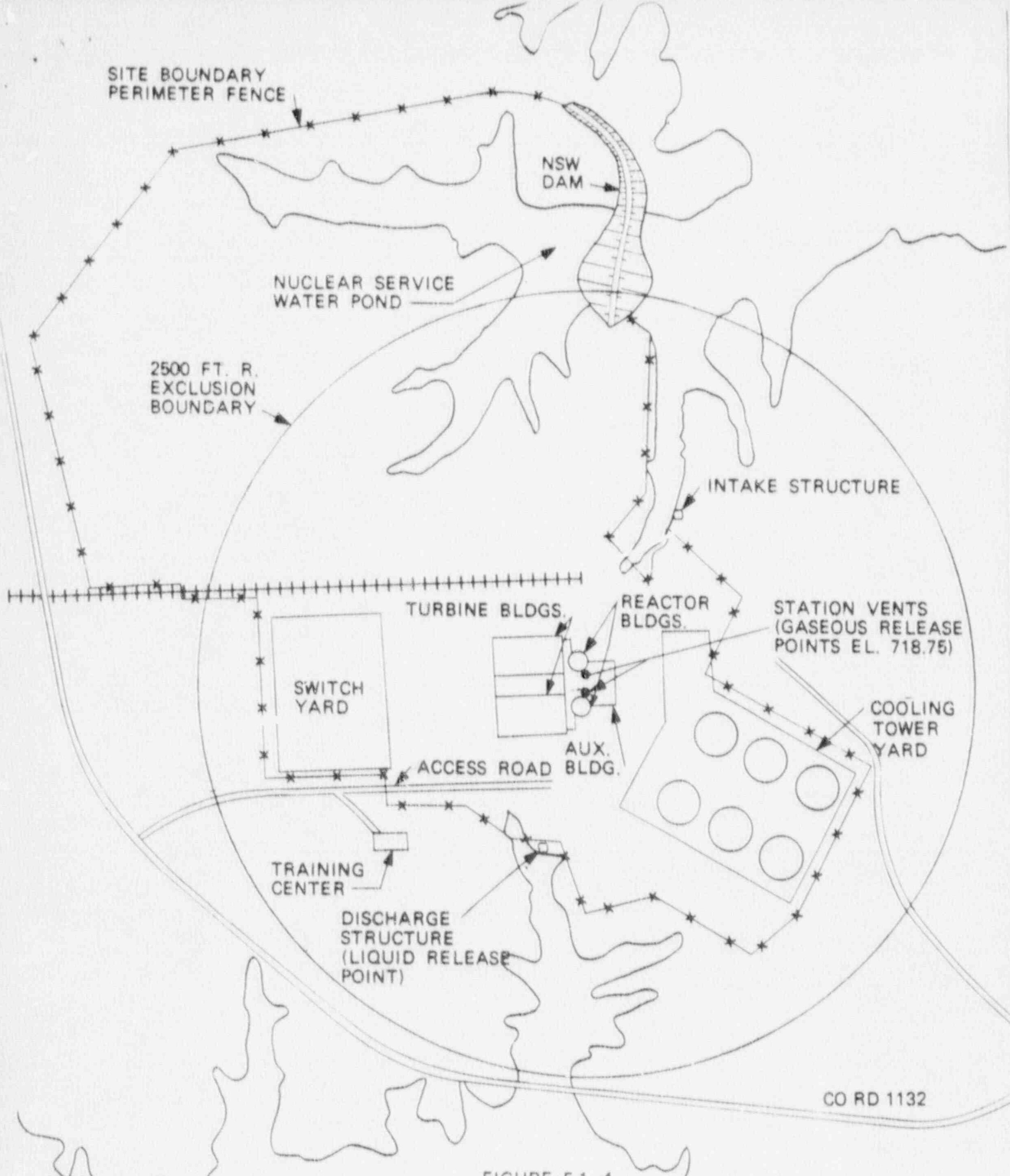

FIGURE 5.1-4

UNRESTRICTED AREA AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS EFFLUENTS

RESPONSIBILITIES (continued)

d. Surveillance of selected plant operations and maintenance activities to provide independent verification* that they are performed correctly and that human errors are reduced to as low as practicable.

e. Investigation of selected unusual events and other occurrences as assigned by Station Management or the Manager of Nuclear Safety Assurance.

AUTHORITY

6.2.3.4 The CSRG shall report to and advise the Manager of Nuclear Safety Assurance, on those areas of responsibility specified in Section 6.2.3.

RECORDS

6.2.3.5 Records of activities performed by the CSRG shall be prepared and maintained for the life of the station. Summary reports of activities performed by the CSRG shall be forwarded each calendar month to the Manager of Nuclear Safety Assurance.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall serve in an advisory capacity to the Shift Supervisor.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Radiation Protection Manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.**

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Manager, Station Training Services 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 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

*Not responsible for sign-off function.

**Except that the experience and other considerations described in Duke Power Company's letters dated August 28, 1985, and July 8, 1986, are acceptable for the six and two applicants for SRO licenses identified therein, respectively.

6.5 REVIEW AND AUDIT

6.5.1 TECHNICAL REVIEW AND CONTROL ACTIVITIES

6.5.1.1 Each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, shall be prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto.

6.5.1.2 Proposed changes to the Appendix A Technical Specifications shall be prepared by a qualified individual/organization. The preparation of each proposed Technical Specification change shall be reviewed by an individual/group other than the individual/group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the Technical Specifications shall be approved by the Station Manager.

6.5.1.3 Proposed modifications to unit nuclear safety-related structures, systems, and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to nuclear safety-related structures, systems, and components shall be approved prior to implementation by the Station Manager; or by the Operating Superintendent, the Technical Services Superintendent, the Maintenance Superintendent, or the Superintendent of Integrated Scheduling, as previously designated by the Station Manager.

6.5.1.4 Individuals responsible for reviews performed in accordance with Specifications 6.5.1.1, 6.5.1.2, and 6.5.1.3 shall be members of the station supervisory staff, previously designated by the Station Manager to perform such reviews. Review of environmental radiological analysis procedures shall be performed by the Corporate System Health Physicist or his designee. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated station review personnel.

6.5.1.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the Station Manager; or by the Operating Superintendent, the Technical Services Superintendent, the Maintenance Superintendent, or the Superintendent of Integrated Scheduling, as previously designated by the Station Manager.

TECHNICAL REVIEW AND CONTROL ACTIVITIES (Continued)

6.5.1.6 All REPORTABLE EVENTS and all violations of Technical Specifications shall be investigated and a report prepared which evaluates the occurrence and which provides recommendations to prevent recurrence. Such reports shall be approved by the Station Manager and transmitted to the Vice President, Nuclear Production, and to the Nuclear Safety Review Board.

6.5.1.7 The Station Manager shall assure the performance of special reviews and investigations, and the preparation and submittal of reports thereon, as requested by the Vice President, Nuclear Production.

6.5.1.8 The station security program, and implementing procedures shall be reviewed at least once per 12 months. Recommended changes shall be approved by the Superintendent of Station Services and transmitted to the Vice President, Nuclear Production, and to the Nuclear Safety Review Board.

6.5.1.9 The station emergency plan, and implementing procedures, shall be reviewed at least once per 12 months. Recommended changes shall be approved by the Station Manager and transmitted to the Vice President, Nuclear Production, and to the Nuclear Safety Review Board.

6.5.1.10 The Station Manager shall assure the performance of a review by a qualified individual/organization of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations, and disposition of the corrective ACTION to prevent recurrence to the Vice President, Nuclear Production and to the Nuclear Safety Review Board.

6.5.1.11 The Station Manager shall assure the performance of a review by a qualifed individual/organization of changes to the PROCESS CONTROL PROGRAM OFFSITE DOSE CALCULATION MANUAL, and Radwaste Treatment Systems.

6.5.1.12 Reports documenting each of the activities performed under Specifications 6.5.1.1 through 6.5.1.11 shall be maintained. Copies shall be provided to the Vice President, Nuclear Production, and the Nuclear Safety Review Board.

6.5.2 NUCLEAR SAFETY REVIEW BOARD (NSRB)

FUNCTION

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

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

AUDITS (Continued)

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety at least once per 6 months;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

e. The Emergency Plan and implementing procedures at least once per 12 months;

f. The Security Plan and implementing procedures at least once per 12 months;

g. The Facility Fire Protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel;

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

i. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months;

j. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months;

k. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months;

l. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months; and

m. Any other area of unit operation considered appropriate by the NSRB or the Vice President, Nuclear Production.

RECORDS

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

a. Minutes of each NSRB meeting shall be prepared, approved, and forwarded to the Vice President, Nuclear Production, and to the Executive Vice President, Power Group within 14 days following each meeting;

RECORDS (Continued)

b. Reports of reviews encompassed by Specification 6.5.2.8 above, shall be prepared, approved, and forwarded to the Vice President, Nuclear Production, and to the Executive Vice President, Power Group within 14 days following completion of the review; and

c. Audit reports encompassed by Specification 6.5.2.9 above, shall be forwarded to the Vice President, Nuclear Production, and to the Executive Vice President, Power Group and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 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 Station Manager; or by (1) Operating Superintendent; (2) Technical Services Superintendent; (3) Maintenance Superintendent; or (4) Superintendent of Integrated Scheduling, as previously designated by the Station Manager, and the results of this review shall be submitted to the NSRB and the Vice President-Nuclear Production.

6.7 SAFETY LIMIT VIOLATION

6.7.1 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 Vice President-Nuclear Production and the NSRB shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Operating Superintendent and Station Manager. 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 NSRB and the Vice President-Nuclear Production within 14 days of the violation; and

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

PROCEDURES AND PROGRAMS (Continued)

e. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1) Training of personnel,

2) Procedures for sampling and analysis, and

3) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to NRC in accordance with 10 CFR 50.4.

STARTUP REPORT

6.9.1.1 A summary report of plant 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 unit.

6.9.1.2 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.

6.9.1.3 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.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the NRC in accordance with 10 CFR 50.4, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
3. Control Bank Insertion Limits for Specification 3/4.1.3.6,
4. Axial Flux Difference Limits, target band, and APL^{ND} for Specification 3/4.2.1,
5. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(Z), W(Z), APL^{ND} and $W(Z)_{BL}$ for Specification 3/4.2.2, and
6. Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $MF_{\Delta H}$, limits for Specification 3/4.2.3.

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

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux

CORE OPERATING LIMITS REPORT (Continued)

Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary).

   (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology.)

3. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987, (W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

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, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC in accordance with 10 CFR 50.4.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10 CFR 50.4 within the time period specified for each report.

6.10 RECORD RETENTION

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

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

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

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

c. All REPORTABLE EVENTS;

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

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

f. Records of radioactive shipments;

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

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

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

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

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

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

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

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

f. Records of reactor tests and experiments;

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