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

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 17
License No. NPF-69

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

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated April 10, 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 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;

   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 amended by 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-69 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 17 are hereby incorporated into this license. Niagara Mohawk Power Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 19, 1990

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 17 TO FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Revise Appendix A as follows:

	Remove Pages	Insert Pages
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DEFINITIONS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

PAGE

ADMINISTRATIVE CONTROLS

REPORTING REQUIREMENTS (Continued)

UNIDENTIFIED LEAKAGE

1.46 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

UNRESTRICTED AREA

1.47 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY, access to which is not controlled by the Niagara Mohawk Power Corporation or the New York State Power Authority for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.48 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered safety features (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.49 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

CORE OPERATING LIMITS REPORT

1.50 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides CORE OPERATING LIMITS for the current operating reload cycle. These cycle-specific CORE OPERATING LIMITS shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these Operating Limits is addressed in individual specifications.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITIONS FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT. The APLHGR Limits shall be reduced as specified in the CORE OPERATING LIMITS REPORT when in single recirculation loop operation.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

3/4.2.2 AVERAGE POWER RANGE MONITOR SETPOINTS

LIMITING CONDITIONS FOR OPERATION

3.2.2 The Average Power Range Monitor (APRM) flow-biased simulated thermal power-upscale scram trip setpoint (S) and flow-biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the relationships specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the APRM flow-biased simulated thermal power-upscale scram trip setpoint and/or the flow-biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint value* within 6 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRACTION OF RATED THERMAL POWER (FRTP) and the CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be determined, the value of T** calculated, and the most recent actual APRM flow-biased simulated thermal power-upscale scram and flow-biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with CMFLPD greater than or equal to FRTP.

d. The provisions of Specification 4.0.4 are not applicable.

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* With CMFLPD greater than the FRTP rather than adjusting the APRM setpoints, the APRM gain may be adjusted so that APRM readings are greater than or equal to 100% times CMFLPD provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

** Definition of T is specified in the CORE OPERATING LIMITS REPORT.

3/4.2.3 MINIMUM CRITICAL POWER RATIO (ODYN OPTION B)

LIMITING CONDITIONS FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT with:

$$\tau = \frac{(\tau_{ave} - \tau_B)}{\tau_A - \tau_B}$$

where:

τ_A = 0.86 seconds, control rod average scram insertion time limit to notch 39 per Specification 3.1.3.3,

$$\tau_B = x + y\left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{1/2}[z],$$

with x, y and z specified in the CORE OPERATING LIMITS REPORT

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i},$$

n = number of surveillance tests performed in cycle

N_i = number of active control rods measured in the i^{th} surveillance test,

τ_i = average scram time to notch 39 of all rods measured in the i^{th} surveillance test

N_1 = total number of active rods measured in Specification 4.1.3.2.a.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

LIMITING CONDITIONS FOR OPERATION

3.2.3 (Continued)

ACTION:

a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within 1 hour, MCPR is determined to be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT.

b. With the main turbine bypass system inoperable per Specification 3.7.7, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within 1 hour, MCPR is determined to be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT.

c. With MCPR less than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT initiate corrective action within 15 minutes to restore MCPR within the required limit. Restore MCPR to within the required limit within 4 hours, if necessary, by reducing THERMAL POWER to the level required.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR shall be determined to be equal to or greater than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT with:

a. τ = 1.0 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2,

   1. At least once per 24 hours,

   2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

   3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR, or

b. τ as defined in Specification 3.2.3 used to determine the limit within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2.

   1. At least once per 24 hours,

   2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

   3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

c. The provisions of Specification 4.0.4 are not applicable.

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITIONS FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.

d. The provisions of Specification 4.0.4 are not applicable.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Rod Block Monitor		
a. Upscale	*	*
b. Inoperative	NA	NA
c. Downscale	≥5% of RATED THERMAL POWER	≥3% of RATED THERMAL POWER
2. APRM		
a. Flow-Biased Neutron Flux - Upscale	*	*
b. Inoperative	NA	NA
c. Downscale	≥4% of RATED THERMAL POWER	≥3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤12% of RATED THERMAL POWER	≤14% of RATED THERMAL POWER
3. Source Range Monitor		
a. Detector Not Full In	NA	NA
b. Upscale	≤1 x 10^5 cps	≤1.6 x 10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥3 cps**	≥1.8 cps**
4. Intermediate Range Monitors		
a. Detector not full in	NA	NA
b. Upscale	≤108/125 divisions of full scale	≤110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥5/125 divisions of full scale	≥3/125 divisions of full scale
5. Scram Discharge Volume		
Water Level - High, Float Switch	≤16.5 in.	≤39.75 in.

TABLE 3.3.6-2 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
6. Reactor Coolant System Recirculation Flow		
a. Upscale	≤108% rated flow	≤111% rated flow
b. Inoperative	NA	NA
c. Comparator	≤10% flow deviation	≤11% flow deviation
7. Reactor Mode Switch		
a. Shutdown Mode	NA	NA
b. Refuel Mode	NA	NA

* Specified in the CORE OPERATING LIMITS REPORT

** For initial loading and startup the count rate may be less than 3 cps if the following conditions are met: the signal to noise ratio is greater than or equal to 20, and the signal is greater than 0.7 cps.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITIONS FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation with:

a. Total core flow greater than or equal to 45% of rated core flow, or
b. THERMAL POWER within the unrestricted zone of Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

   1. Within four hours:

      a) Place the recirculation flow control system in the Loop Manual (Position Control) mode, and

      b) Reduce THERMAL POWER to ≤ 70% of RATED THERMAL POWER, and,

      c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.08 per Specification 2.1.2, and,

      d) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit per Specification 3.2.1, and,

      e) Reduce the Average Power Range Monitor (APRM) Scram and Rod Block and Rod Block Monitor Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specifications 2.2.1, 3.2.2 and 3.3.6.

      f) Reduce the volumetric drive flow rate of the operating recirculation loop to ≤ 41,800** gpm.

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* See Special Test Exception 3.10.4.

** This value represents the volumetric recirculation loop drive flow which produces 100% core flow at 100% THERMAL POWER.

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming an LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure-dependent steady-state gap conductance and rod-to-rod local peaking factor. The limiting value for APLHGR is specified in the CORE OPERATING LIMITS REPORT for two-recirculation-loop operation.

The calculational procedure used to establish the APLHGR specified in the CORE OPERATING LIMITS REPORT is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B3.2.1-1.

For plant operations with single recirculation loop the MAPLHGR limits are specified in the CORE OPERATING LIMITS REPORT. The constant factor is derived from LOCA analyses initiated from single loop operation to account for earlier boiling transition at the limiting fuel node compared to the standard LOCA evaluations.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow-biased simulated thermal power-upscale scram setting and flow-biased neutron flux upscale control rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than the fuel cladding integrity safety limit or that greater than or equal to 1% plastic strain does not occur in the degraded situation. The scram setpoint and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and CMFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady-state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady-state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient, assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0.3 that are input to a GE-core dynamic behavior transient computer program. The code used to evaluate pressurization events is described in NEDO-24154 (Reference 3) and the program used in non-pressurization events is described in NEDO-10802 (Reference 2). The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic TASC code described in NEDE-25149 (Reference 4). The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the K_f factor specified in the CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow, the required MCPR is the product of the MCPR and the K_f factor. The K_f factors assure that the Safety Limit MCPR will not be violated. The K_f factors are calculated as described in Reference 5.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial startup testing of the plant, an MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts,

3/4.2.3 MINIMUM CRITICAL POWER RATIO

3/4 2.3 (Continued)

while still allotting time for the power distribution to stabilize. The requirement for calculating MCPR after initially determining that a LIMITING CONTROL ROD PATTERN exists ensures MCPR will be known following a change in THERMAL POWER or power shape, and therefore avoid operation while exceeding a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the linear heat generation rate (LHGR) in any rod is less than the design linear heat generation rate even if fuel pellet densification is postulated. The daily requirement for calculating LHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient, since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate LHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating LHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that LHGR will be known following a change in THERMAL POWER or power shape that could place operation exceeding a thermal limit.

References

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, latest approved revision.

2. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, NEDO-10802, latest approved revision.

3. Qualification of the One Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154, latest approved revision.

4. TASC 01-A Computer Program for the Transient Analysis of a Single Channel, Technical Description, NEDE-25149, latest approved revision.

5. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-9-US latest approved revision.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.8 (Continued)

The Semiannual 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), pursuant to Specifications 6.13 and 6.14, respectively, as well as any major change to liquid, gaseous, or solid radwaste treatment systems pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation of why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.7.9 or 3.3.7.10, respectively, and a description of the events leading to liquid holdup tanks exceeding the limits of Specification 3.11.1.4.

CORE OPERATING LIMITS REPORT

6.9.1.9

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle for the following:

   1) The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1.

   2) The Average Power Range Monitor (APRM) flow-biased simulated thermal power-upscale scram trip setpoint and flow-biased neutron flux-upscale control rod block trip setpoint for Specification 3.2.2.

   3) The K_f core flow adjustment factor for Specification 3.2.3.

   4) The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.3.

   5) The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.4.

   6) Control Rod Block Instrumentation Setpoints for the rod block monitor upscale and APRM flow biased neutron flux upscale trip and allowable values for Specification 3.3.6.

   and shall be documented in the CORE OPERATING LIMITS REPORT.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC; specifically those described in the following document.

   1) General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566 latest approved revision.

CORE OPERATING LIMITS REPORT

6.9.1.9 (Continued)

2) General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-9-US latest approved revision.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2. Special reports shall be submitted 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, of the Code of Federal Regulations (10 CFR), the following records shall be retained for at least the minimum period indicated.

6.10.1.1 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 submitted to the Commission

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

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

RECORD RETENTION

6.10.1.2 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 5.7.1-1

f. Records of reactor tests and experiments

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

h. Records of inservice inspections performed pursuant to these Technical Specifications

i. Records of quality assurance activities required by the Operational Quality Assurance Manual, and not listed in Specification 6.10.1.1

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 SORC and the SRAB

l. Records of the service lives of all snubbers, including the date at which the service life commences and associated installation and maintenance records

m. Records of analyses required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed

n. Records of unit radiation and contamination surveys

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by 10 CFR 20.203 (c)(2), 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 requiring issuance of a Radiation Work Permit (RWP)**. 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 which continuously indicates the radiation dose rate in the area

b. A radiation monitoring device which 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

c. An individual qualified in radiation protection (i.e., 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 by the Radiation Protection Supervisor or the Radiation Protection Supervisor's designee in the RWP

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem* shall be provided with locked doors to prevent unauthorized entry, and the keyed access shall be maintained under the administrative control of the Station Shift Supervisor or the designee on duty and/or the Radiation Protection Supervisor or designee. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with such radiation levels that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrem* that are located within large areas, such as the drywell, where no enclosure exists that can be locked, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off,

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*Measurements made at 18 inches from the source of radioactivity.

**Health physics personnel or personnel escorted by health physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA

6.12.2 (Continued)

conspicuously posted, and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, continuous surveillance, direct or remote (such as use of closed circuit TV cameras), may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM

6.13.1 The PROCESS CONTROL PROGRAM (PCP) shall be approved by the Commission before implementation.

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   2. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   3. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

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

6.14 OFFSITE DOSE CALCULATION MANUAL

6.14.1 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be approved by the Commission before implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed; each page should be numbered, dated, and marked with the revision number; appropriate analyses or evaluations justifying the change(s) should be included;

OFFSITE DOSE CALCULATION MANUAL

6.14.2 (Continued)

2. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

3. Documentation of the fact that the SORC has reviewed the change and found it acceptable.

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

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

6.15.1 Licensee-initiated major changes to the radwaste treatment systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual 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 that precedes the time when the change is to be made;

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

*Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

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

6.15.1.a (Continued)

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.