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

ILLINOIS POWER COMPANY, ET AL

DOCKET NO. 50-461

CLINTON POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 28
License No. NPF-62

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

   A. The application for amendment by Illinois Power Company* (IP), and Soyland Power Cooperative, Inc., (the licensees) dated June 12, 1989 as amended August 17, 1989, 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-62 is hereby amended to read as follows:

*Illinois Power Company is authorized to act as agent for Soyland Power Cooperative, Inc. and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 28 , are hereby incorporated into this license. Illinois 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

John W. Craig

John W. Craig, Project Director
Project Directorate III-2
Division of Reactor Projects - III, IV, V and Special Projects

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 22, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 28

FACILITY OPERATING LICENSE NO. NPF-62

DOCKET NO. 50-461

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove	Insert
i	i
v	v
xxv	xxv
1-2	1-2
1-3	1-3
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-4a	-
3/4 2-4b	-
3/4 2-4c	-
3/4 2-4D	-
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 4-1	3/4 4-1
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4
B 3/4 4-1	B 3/4 4-1
6-21	6-21
-	6-21a

1.0 DEFINITIONS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

ADMINISTRATIVE CONTROLS

CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM RESPONSE TIME

1.7 The CONTAINMENT AND REACTOR VESSEL ISOLATION AND CONTROL SYSTEM (CRVICS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

CORE ALTERATION

1.8 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of the SRMs, IRMs, or TIPs, or special movable detectors, is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the Clinton-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.

CRITICAL POWER RATIO

1.10 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of an approved General Electric Critical Power correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

DRYWELL INTEGRITY

1.12 DRYWELL INTEGRITY shall exist when:

a. All drywell penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE drywell automatic isolation system or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.4-1 of Specification 3.6.4.

b. The drywell equipment hatch is closed and sealed.

c. The drywell airlock is OPERABLE pursuant to Specification 3.6.2.3.

DRYWELL INTEGRITY (Continued)

d. The drywell leakage rates are within the limits of Specification 3.6.2.2.

e. The suppression pool is OPERABLE pursuant to Specification 3.6.3.1.

f. The sealing mechanism associated with each drywell penetration, e.g., welds, bellows or O-rings, is OPERABLE.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.13 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.14 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function; i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.15 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker from initial movement of the associated:

a. Turbine stop valves and
b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

1.16 [DELETED]

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION 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.

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 required limits:

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,

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

d. The provisions of Specification 4.0.4 are not applicable.

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Figure 3.2.1-1 DELETED.

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Figure 3.2.1-2 DELETED.

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Figure 3.2.1-3 DELETED.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the limits 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:

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

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR shall be determined to be equal to or greater 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,

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

d. The provisions of Specification 4.0.4 are not applicable.

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Figure 3.2.3-1 DELETED.

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3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION 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.

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION 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, or

c. THERMAL POWER within the restricted zone† of Figure 3.4.1.1-1 and APRM or LPRM†† noise levels not larger than three times their established baseline noise levels.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

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

   1. Within 4 hours:

      a) Place the recirculation flow control system in the Local 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 the CORE OPERATING LIMITS REPORT, and

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

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*See Special Test Exception 3.10.4.

†The operating region for which monitoring is required. See Surveillance Requirement 4.4.1.1.2.

††Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

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 a 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 Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is the MAPLHGR.

The MAPLHGR limits specified in the CORE OPERATING LIMITS REPORT are multiplied by the smaller of the flow-dependent MAPLHGR factor ($MAPFAC_f$) or the power-dependent MAPLHGR factor ($MAPFAC_p$) corresponding to existing core flow and power conditions to assure the adherence to fuel mechanical design bases during the most limiting transient (Reference 2). The $MAPFAC_F$ factors are determined using the three-dimensional BWR simulator code to analyze slow flow runout transients. The maximum runout flow settings of 102.5% and 109% include design allowances for recirculation flow instrument uncertainties (2.5% and 2.0% respectively) to ensure that the rated flow conditions of 100% and 107% can be achieved. The $MAPFAC_P$ factors are generated using the same data base as the $MCPR_p$ to protect the core from plant transients other than core flow runout.

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. Differences in this analysis compared to previous analyses can be broken down as follows.

a. Input Changes

1. Corrected Vaporization Calculation - Coefficients in the vaporization correlation used in the REFLOOD code were corrected.

2. Incorporated more accurate bypass areas - The bypass areas in the top guide were recalculated using a more accurate technique.

3. Corrected guide tube thermal resistance.

4. Correct heat capacity of reactor internals heat nodes.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

2. Incorporate NRC pressure transfer assumption - The assumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

1. Break Areas - The DBA break area was calculated more accurately.

b. Model Change

1. Improved Radiation and Conduction Calculation - Incorporation of CHASTE 05 for heatup calculation.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

For plant operation with a single recirculation loop, the MAPLHGR limits specified in the CORE OPERATING LIMITS REPORT are multiplied by the smallest of $MAPFAC_f$, $MAPFAC_p$ or 0.85 (Reference 2). The constant factor, 0.85, is derived from LOCA analyses initiated from single loop operation to account for earlier boiling transition at the limiting fuel node compared to standard LOCA evaluations.

3/4.2.2 APRM SETPOINTS [DELETED]

BASES TABLE B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS*

Plant Parameters:

Core THERMAL POWER .... 3015 MWt** which corresponds to 105% of rated steam flow

Vessel Steam Output .... 13.08×10^6 lb_m/hr which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure............. 1060 psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks 2.2 ft^2.

b. Small Breaks 0.09 ft^2.

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kW/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial and Reload Cores	8 x 8	#	1.4	1.17***

*A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and Section 6.3 of the FSAR.

**This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

***For single recirculation loop operation, loss of nucleate boiling is assumed at 0.1 seconds after a LOCA, regardless of initial MCPR. For core flows less than 85% of rated, the initial MCPR is taken from the $MCPR_f$ Curve specified in the CORE OPERATING LIMITS REPORT.

#This value is specified in the CORE OPERATING LIMITS REPORT.

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 in Specification 2.1.2, 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 power-flow maps of Figures B 3/4.2.3-1 or B 3/4.2.3-2 give operational limits for double or single recirculation loop operation, respectively.

The evaluation of a given transient begins with the system initial parameters identified in Reference 3 that are input to a GE-core dynamic behavior transient computer program. The codes used to evaluate pressurization and non-pressurization events are described in Reference 3. The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the $MCPR_f$ and $MCPR_p$ specified in the CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

The $MCPR_f$s were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along the most limiting power flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. The maximum runout flow settings (109% and 102.5%) include design allowances for recirculation flow instrument uncertainties (2% and 2.5% respectively) to ensure that the rated flow conditions (107% and 100%) can be achieved. Using this relative bundle power, the MCPRs were calculated at different points along the most limiting power flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

*The values associated with this limit are specified in the CORE OPERATING LIMITS REPORT.

3/4.4.1 RECIRCULATION SYSTEM

The impact of single recirculation loop operation upon plant safety is assessed and shows that single-loop operation is permitted if the MCPR fuel cladding safety limit is increased as noted by Specification 2.1.2, APRM scram and control rod block setpoints are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2, respectively, MAPLHGR limits are decreased in accordance with the values specified in the CORE OPERATING LIMITS REPORT, and MCPR operating limits are adjusted in accordance with the values specified in the CORE OPERATING LIMITS REPORT.

Additionally, surveillance on the volumetric flow rate of the operating recirculation loop is imposed to exclude the possibility of excessive core internals vibration. The surveillance on differential temperatures below (30%)* THERMAL POWER or (50%)* rated recirculation loop flow is to mitigate the undue thermal stress on vessel nozzles, recirculation pump, and vessel bottom head during the extended operation of the single recirculation loop mode.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation. Significant degradation is indicated if more than one of three specified surveillances performed confirms unacceptable deviations from established patterns or relationships. The surveillances, including the associated acceptance criteria, are in accordance with General Electric Service Information Letter No. 330, the recommendations of which are considered acceptable for verifying jet pump operability according to NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure." Performance of the specified surveillances, however, is not required when thermal power is less than 25% RATED THERMAL POWER because flow oscillations and jet noise precludes the collection of repeatable meaningful data during low flow conditions approaching the threshold response of the associated flow instrumentation.

Recirculation loop flow mismatch limits are in compliance with ECCS LOCA analysis design criteria for two recirculation loop operation. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. In the case where the mismatch limits cannot be maintained during two loop operation, continued operation is permitted in a single recirculation loop mode.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Sudden equilization of a temperature difference > 100°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

*Initial Values. Final values to be determined during Startup Testing based on the threshold THERMAL POWER and recirculation loop flow which will sweep the cold water from the vessel bottom head preventing stratification.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual 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 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 changes to liquid, gaseous, or Solid Radwaste Treatment Systems pursuant to Specification 6.15. It will 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.

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam system safety/relief valves, shall be submitted on a monthly basis to the Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, 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. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in General Electric Standard Application for Reactor Fuel (GESTAR), NEDE-24011-P-A-8, as amended (latest approved version) and Maximum Extended Operating Domain and Feedwater Heater Out-of-Service Analysis for Clinton Power Station, NEDC-31546P, August 1988. The core operating limits shall be determined so that all application 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, for each reload cycle, shall be submitted upon issuance 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 to the Regional Administrator of the Regional Office of the NRC 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.

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

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

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

c. All REPORTABLE EVENTS.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications and the Fire Protection Program.