December 1, 1987

Dockets Nos.: 50-321
and 50-366

SEE PROPOSED CHANGES TO TECH. SPECS.

Mr. James P. O'Reilly
Senior Vice President - Nuclear Operations
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. O'Reilly:

Subject: Issuance of Amendment Nos.149 and 86 to Facility Operating Licenses DPR-57 and NPF-5 - Edwin I. Hatch Nuclear Plant, Units 1 and 2 (TACS 55713/55714)

The Commission has issued the enclosed Amendments Nos. 149 and 86 to Facility Operating Licenses DPR-57 and NPF-5, for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated February 13, 1987.

The amendments modify the Technical Specifications to incorporate the revised reporting requirements of 10 CFR 50.72 and 50.73 and the revised reporting requirements for primary coolant iodine spiking and remove existing requirements for plant shutdown if primary coolant iodine activity limits are exceeded for 800 hours within a 12-month period.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/S/

Lawrence P. Crocker, Project Manager
Project Directorate II-3
Division of Reactor Projects-I/II

Enclosures:
1. Amendment No. 149 to DPR-57
2. Amendment No. 86 to NPF-5
3. Safety Evaluation

cc w/enclosures:
See next page

	LPC	KNJ
PD#II-3/DRP-I/II	PD#II-3/DRP-I/II	PD#II-3/DRP-I/II
MDuncan/rad	LCrocker	K. JABBOUR
06/1/87	~~06~~ 7/1/87	12/01/87

8712220160 871201
PDR ADOCK 05000321
P PDR

Mr. James P. O'Reilly
Georgia Power Company

Edwin I. Hatch Nuclear Plant,
Units Nos. 1 and 2

cc:
G. F. Trowbridge, Esq.
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, D.C. 20036

Mr. L. T. Gucwa
Engineering Department
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Nuclear Safety and Compliance Manager
Edwin I. Hatch Nuclear Plant
Georgia Power Company
P. O. Box 442
Baxley, Georgia 31513

Mr. Louis B. Long
Southern Company Services, Inc.
P. O. Box 2625
Birmingham, Alabama 35202

Resident Inspector
U.S. Nuclear Regulatory Commission
Route 1, P. O. Box 279
Baxley, Georgia 31513

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, Suite 2900
Atlanta, Georiga 30323

Mr. Charles H. Badger
Office of Planning and Budget
Room 610
270 Washington Street, S.W.
Atlanta, Georgia 30334

Mr. J. Leonard Ledbetter, Commissioner
Department of Natural Resources
270 Washington Street, N.W.
Atlanta, Georgia 30334

Chairman
Appling County Commissioners
County Courthouse
Baxley, Georgia 31513

DATED December 1, 1987

AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE DPR-57, EDWIN I. HATCH, UNITS 1 & 2
AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NPF-05, EDWIN I. HATCH, UNITS 1 & 2

DISTRIBUTION:
[redacted]
NRC PDR
Local PDR
PRC System
PD#II-3 Reading
M. Duncan
L. Crocker
B. J. Youngblood
D. Hagan
T. Barnhart (8)
W. Jones
ACRS (10)
OGC-Bethesda
S. Varga/G. Lainas
U. Cheh
ARM/LFMB
GPA/PA
E. Butcher
L. Reyes

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 149
License No. DPR-57

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 1 (the facility) Facility Operating License No. DPR-57 filed by Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, (the licensee) dated February 13, 1987, 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 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.

8712220165 871201
PDR ADOCK 05000321
P PDR

REGULATORY DOCKET FILE COPY

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. DPR-57 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 149, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Kahtan N. Jabbour, Acting Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 1, 1987

9/2/87

7/24/87

PD#II-3/DRP-I/II	PD#II-3/DRP-I/II	OGC-Bethesda	PD#II-3/DRP-I/II
MDuncan/rad	LCrocker	M. Karman	KJabbour, Acting PD
06/ 1 /87	~~06~~ 7/ 1 /87	09/ 2 /87	12/01 /87

ATTACHMENT TO LICENSE AMENDMENT NO. 149

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

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

Remove Page	Insert Page
vi	vi
1.0-10	1.0-10
3.6-4	3.6-4
3.6-5	3.6-5
3.6-9	3.6-9
3.7-6	3.7-6
3.7-6a	3.7.6a
3.11-2a	3.11-2a
3.11-5	3.11-5
6-7	6-7
6-10	6-10
6-12	6-12
6-13	6-13
6-14	6-14
6-15	6-15
6-15b	6-15b
6-15d	6-15d
6-16	6-16
6-17	6-17
6-18	6-18

Replace the following pages of the Appendix B Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

4-1	4-1
4-2	4-2
5-8	5-8

EEE. MILK ANIMAL

A cow or goat that is producing milk for human consumption.

FFF. DOSE EQUIVALENT IODINE

The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/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 or those in NRC Regulatory Guide 1.109, Revision 1, October 1977.

GGG. ACTION

ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.

HHH. CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

III. STAGGERED TEST BASIS

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or other designated components at the beginning of each subinterval.

JJJ. REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

3.6.F. Reactor Coolant Chemistry

1. Radioactivity

Whenever the reactor is critical the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 0.2 μCi/gm of dose equivalent* I-131.

If activity concentration >0.2 μ Ci/gm dose equivalent I-131 but ≤ 4.0 μ Ci/gm for more than 48 hours during one continuous time interval, or >4.0 μ Ci/gm, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

*That I-131 concentration which alone would produce the same thyroid dose as the quantity and iodine mixture actually present.

4.6.F. Reactor Coolant Chemistry

1. Radioactivity

a. During equilibrium power operation an isotopic analysis, including quantitative measurements for at least I-131, I-132, I-133, and I-135 shall be performed monthly on a coolant liquid sample.

b. During equilibrium power operation an isotopic analysis, including quantitative measurements for at least Xe-133 and Xe-135 shall be performed monthly on a steam jet air ejector off-gas sample.

c. Additional coolant samples shall be taken whenever the reactor coolant dose equivalent I-131 concentration exceeds 0.2 μ Ci/gm and any of the following conditions are met:

1) During startup

2) Following a power change exceeding 15% of rated thermal power in less than 1 hour (net change averaged for 1 hour).

3) The off-gas level, at the SJAE, increases by more than 10,000 μCi/sec in 1 hour at release rate ≤ 75,000 μCi/sec, or

4) The off-gas level at the SJAE, increases by more than 15% in 1 hour at release rate > 75,000 μCi/sec.

5) Whenever the reactor coolant dose equivalent I-131 concentration exceeds 4.0 μCi/gm.

The first additional coolant sample shall be taken between 2 and 6 hours following the change in thermal power or off-gas level. Additional coolant liquid samples shall be taken at 4-hour intervals for 48 hours, or until a stable iodine concentration below the limiting value of 4.0 μCi/gm is established. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration. If the total iodine activity of the sample is below 0.2 μCi/gm, an isotopic analysis to determine equivalent I-131 is not required.

All data obtained from normal and any additional samples shall be included in the annual report.

3.6.H.1. Relief/Safety Valves

a. When one or more relief/safety valve(s) is known to be failed*** an orderly shutdown shall be initiated and the reactor depressurized to less than 113 psig within 24 hours. Prior to reactor startup from a cold condition all relief/safety valves shall be operable.**

b. With one or more relief/safety valve(s) stuck open, place the reactor mode switch in the shutdown position.

c. With one or more safety/relief valve tailpipe pressure switches of a safety/relief valve declared inoperable and the associated safety/relief valve(s) otherwise indicated to be open, place the reactor mode switch in the Shutdown position.

d. With one safety/relief valve tailpipe pressure switch of a safety/relief valve declared inoperable and the associated safety/relief valve(s) otherwise indicated to be closed, plant operation may continue. Remove the function of that pressure switch from the low low set logic circuitry until the next COLD SHUTDOWN. Upon COLD SHUTDOWN, restore the pressure switch(es) to OPERABLE status before STARTUP.

e. With both safety/relief valve tailpipe pressure switches of a safety/relief valve declared inoperable and the associated safety/relief valve(s) otherwise indicated to be closed, restore at least one inoperable switch to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6.H.1. Relief/Safety Valves

a. End of Operating Cycle

Approximately one-half of all relief/safety valves shall be benchchecked or replaced with a benchchecked valve each refueling outage. All 11 valves will have been checked or replaced upon the completion of every second operating cycle.

b. Each Operating Cycle

Once during each operating cycle, at a reactor pressure > 100 psig each relief valve shall be manually opened until thermocouples downstream of the valve indicate steam is flowing from the valve.

c. Integrity of Relief Valve Bellows*

The integrity of the relief valve bellows shall be continuously monitored and the pressure switch calibrated once per operating cycle and the accumulators and air piping shall be inspected for leakage once per operating cycle.

d. Relief Valve Maintenance

At least one relief valve shall be disassembled and inspected each operating cycle.

e. Operability of Tailpipe Pressure Switches

The tailpipe pressure switch of each relief/safety valve shall be demonstrated operable by performance of a:

1. Functional Test:

   a. At least once per 31 days, except that all portions of instrumentation inside the primary containment may be excluded from the functional test, and

*Does not apply to two-stage Target Rock SRVs

**The Reflief/Safety valves are not required to be operable for performance of inservice hydrostatic or pressure testing with reactor pressure greater than 113 psig and all control rods inserted. Overpressure protection will be provided as required by ASME Code.

***The failure or malfunction of any safety/relief valve shall be reported by telephone within 24 hours; confirmed by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office or his designee no later than the first working day following the event; and a written follow-up report within 30 days. The written follow-up report should be completed in accordance with 10 CFR 50.73 or other applicable requirements.

4.7.A.2.e. Type B Test - Leak Tests of Penetrations with Seals and Bellows (Continued) (Tables 3.7-2 and 3.7-3)

(1) Primary containment components which seal or penetrate the pressure containing boundary of the containment shall be tested at a pressure not less than P_a. These components shall be tested at each major refueling shutdown or at intervals not to exceed two years.

(2) (a) The personnel air lock shall be tested at intervals not to exceed six months at P_a by pressurizing the compartment between the two air lock doors.

During intervals of door use when containment integrity is required, the door seals shall be tested at 10 psig after each opening.

(b) Personnel air lock leakage shall not exceed 0.05 L_a.

f. Type C Tests-Local Leak Tests of Containment Isolation Valves (Tables 3.7-1 and 3.7-4)

Type C tests shall be performed under the program established in Appendix J of 10 CFR Part 50.

Containment isolation valves (except for main steam line isolation valves) shall be tested at a pressure not less than P_a. Type C tests shall be performed at each major refueling shutdown or at intervals not to exceed two years.*

*All Type B and Type C Leakage Tests (i.e., Local Leak Rate Tests) that fail (i.e., test leakage is such that an LER would be required) during an outage shall be reported according to 10 CFR 50.73 by one, 30-day written report that is due within 30 days of the first leakage test failure in the outage. All other leakage test failures discovered during the outage will be reported in a revision to the original report due within 30 days after the end of the outage.

g. Acceptance Criteria for Type B and Type C Tests

The combined leakage rate of components subject to Type B and C tests shall be determined under the program established in Appendix J of 10 CFR Part 50 and shall not exceed 0.6 L_a.*

h. Main Steam Line Isolation Valves

The main steam line isolation valves shall be tested at a pressure of 28 psig for leakage at least once per operating cycle. If a total leak rate of 11.5 scf per hour for any one main steam line isolation valve is exceeded, repairs and retest shall be performed to correct this condition.

*All Type B and Type C Leakage Tests (i.e., Local Leak Rate Tests) that fail (i.e., test leakage is such that an LER would be required) during an outage shall be reported according to 10 CFR 50.73 by one, 30-day written report that is due within 30 days of the first leakage test failure in the outage. All other leakage test failures discovered during the outage will be reported in a revision to the original report due within 30 days after the end of the outage.

3.11.C. Minimum Critical Power Ratio (MCPR)

For single-loop operation, the MCPR limit is increased by 0.01 over the comparable two-loop value.

If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, then reduce reactor power to less than 25% of rated thermal power within the next four(4) hours. If the Limiting Condition for Operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required.

3.11.C. Minimum Critical Power Ratio (MCPR) (Continued)

At core thermal power levels less than or equal to 25%, 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 and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at the 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluations below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% 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 when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

MEETING FREQUENCY

6.5.1.4 The PRB shall meet at least once per calendar month and as convened by the PRB Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the PRB necessary for the performance of the PRB responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and three voting members including alternates.

RESPONSIBILITIES

6.5.1.6 The Plant Review Board shall be responsible for:

a. Review of all procedures required by Specification 6.8 and changes thereto, except those for the Radiological Environmental Monitoring Program, any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

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

c. Review of all proposed changes to Appendix "A" Technical Specifications.

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

e. Investigation of all reportable violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Vice President-Plant Hatch, the Senior Vice President-Nuclear Operations, and to the Safety Review Board (SRB).

f. Review of all REPORTABLE EVENTS.

g. Review of unit operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Plant Manager or the SRB.

QUORUM

6.5.2.6. The minimum quorum of the SRB necessary for the performance of the SRB review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least a majority of the members. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7. The SRB shall be responsible for the review of:

a. The safety evaluations for (1) changes to procedures, equipment or systems and (2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. All REPORTABLE EVENTS.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meetings minutes of the Plant Review Board.

k. The Radiological Environmental Monitoring Program and the results thereof annually.

l. The Offsite Dose Calculation Manual, Process Control Program, and implementing procedures at least once per 24 months.

AUTHORITY

6.5.2.9. The SRB shall report to and advise the Senior Vice President - Nuclear Operations on those areas of responsibility specified in Sections 6.5.2.7. and 6.5.2.8.

RECORDS

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

a. Minutes of each SRB meeting shall be prepared, approved and forwarded to the Senior Vice President-Nuclear Operations within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7. above, shall be prepared, approved and forwarded to the Senior Vice President-Nuclear Operations within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8. above, shall be forwarded to the Senior Executive Vice President, the Senior Vice President-Nuclear Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6. REPORTABLE EVENT ACTION

6.6.1. The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and/or 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 PRB and the results of this review shall be submitted to the SRB, the Vice President-Plant Hatch, and the Senior Vice President-Nuclear Operations.

6.7. SAFETY LIMIT VIOLATION

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

a. The unit shall be placed in at least HOT SHUTDOWN within two hours.

b. The Safety Limit violation shall be reported to the Commission as soon as practical and in all cases within one hour of occurrence. The Vice President-Plant Hatch, the Senior Vice President-Nuclear Operations and the SRB shall be notified within 24 hours.

SAFETY LIMIT VIOLATION (Continued)

c. A Licensee Event Report shall be prepared pursuant to 10 CFR 50.73.

d. The Licensee Event Report shall be submitted to the Commission in accordance with 10 CFR 50.73, and to the PRB, the SRB, the Vice President-Plant Hatch, and the Senior Vice President-Nuclear Operations within 30 days of the violation.

6.8. PROCEDURES

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

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

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

6.8.2 Each procedure of 6.8.1. and other procedures which the Plant Manager or Plant Support Manager has determined to affect nuclear safety, and changes thereto, shall be reviewed by the PRB and approved by the appropriate member of plant management, designated by the Plant Manager or Plant Support Manager prior to implementation. The Plant Manager or Plant Support Manager will approve administrative procedures, security plan implementing procedures, and changes thereto. The Manager-Plant Training and Onsite Emergency Preparedness shall approve the emergency plan implementing procedures and changes thereto. All other procedures of this specification and changes thereto will be approved by the department head designated by the Plant Manager or Plant Support Manager. The procedures of this specification shall be reviewed periodically as set forth in administrative procedures.

6.8.3. Temporary changes to procedures of 6.8.1. above may be made provided:

a. The intent of the original procedure is not altered.

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 the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

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 plant.

6.9.1.2. The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. 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 three months until all three events have been completed.

ANNUAL REPORTS[1]

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

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

ANNUAL REPORTS (Continued)

6.9.1.5. Reports required on an annual basis shall include:

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

b. Documentation of all challenges to safety/relief valves.

c. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.6.F.1. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

d. Any other unit unique reports required on an annual basis.

ANNUAL RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE REPORT[(a)]

6.9.1.6 Routine radiological environmental surveillance reports covering the radiological environmental surveillance activities related to the plant during the previous calendar year shall be submitted prior to May 1 of each year. A single report may fulfill this requirement for both units.

6.9.1.7 The Annual Radiological Environmental Surveillance Report shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the reporting period, including (as appropriate) a comparison with the preoperational studies, operational controls, previous environmental surveillance reports, and an assessment of any observed impacts of the plant operation on the environment. The reports shall also include the

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

[a]This tabulation supplements the requirements of 20.407 of 10 CFR Part 20.

TABLE 6.9.1.7-1

ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM SUMMARY

Name of Facility Edwin I. Hatch Nuclear Plant Docket No. 50-321, 50-366

Location of Facility Appling County, Georgia Reporting Period ________

				Location of Highest Annual Mean			
Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection[a]	All Indicator Locations Mean Range	Name, Distance, and Direction	Mean Range[b]	Control Locations Mean Range	Number of REPORTABLE EVENTS

a. Lower Limit of Detection is defined in table notation a. of table 4.16.1-1, Specification 4.16.1. of Unit 1.

b. Mean and range based upon detectable measurements only. Fraction of detectable measurements at specified locations is indicated in parentheses.

e. Type of container, e.g., LSA, type A, type B, large quantity.

f. Solidification agent, e.g., cement.

The Radioactive Effluent Release Report shall include (on a quarterly basis) unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents that were in excess of 1 Ci, excluding dissolved and entrained gases and tritium for liquid effluents, or those in excess of 150 Ci of noble gases or 0.02 Ci of radioiodines for gaseous releases.

The Radioactive Effluent Release Report shall include any changes to the PROCESS CONTROL PROGRAM and to the OFFSITE DOSE CALCULATION MANUAL made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the Regional Office of Inspection and Enforcement no later than the 15th of each month following the calendar month covered by the report.

(This page is intentionally left blank.)

(This page is intentionally left blank.)

SPECIAL REPORTS

6.9.2. Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified and for each activity shown in Table 6.9.2-1. Special reports for fire protection equipment operating and surveillance requirements shall be submitted, as required, by the Fire Hazards Analysis and its Appendix B requirements.

6.10. RECORD RETENTION

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.

6.10.1. The following records shall be retained for at least five 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.

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.

4.0 Special Surveillance and Study Activities

4.1 Erosion Control Inspection

4.2 Unusual or Important Events Requirements

Requirements

The licensee shall be alert to the occurrence of unusual or important events. Unusual or important events are those that cause potentially significant environmental impact, or could be of public interest concerning environmental impact from plant operation. The following are examples: unusual or important bird impaction events on cooling tower structures or meteorological towers, onsite plant or animal disease outbreaks, unusual mortality of any species protected by the Endangered Species Act of 1973, fish kills near the HNP site, and significant violations of relevant permits and certifications.

Action

Should an unusual or important event occur, the licensee shall make a report to the NRC as required by 10 CFR 50.72 or 10 CFR 50.73.

Bases

Prompt reporting to the NRC of unusual or important events, as described, is necessary for responsible and orderly regulation of the nation's system of nuclear power reactors. The information thus provided may be useful or necessary to others concerned with the same environmental resources. Prompt knowledge and action may serve to alleviate the magnitude of environmental impact or to place it into a perspective broader than that available to the licensee. The NRC also has an obligation to be responsible to inquiries from the public and the news media concerning potentially significant environmental events at nuclear power stations.

4.3 Exceeding Limits of Other Relevant Permits

Requirements

The licensee shall notify the NRC of occurrences exceeding the limits specified in relevant permits and certificates issued by other Federal, State, and local agencies that are reportable to the agency that issued the permit. This requirement shall apply only to topics of NEPA concern within the NRC area of responsibility as identified in the Environmental Technical Specifications (ETS).

This requirement shall commence with the date of issuance of the operating license for Unit 2 and continue until approval for modification or termination is obtained from the NRC in accordance with section 5.6.3.

Action

The licensee shall make a report to the NRC in the event of a REPORTABLE EVENT exceeding a limit specified in a relevant permit or certificate issued by another Federal, State, or local agency. The report shall be submitted within the time limit specified by the reporting requirement of the corresponding certification or permit issued pursuant to Section 401 or 402 of PL 92-500. The report will consist of a copy of the report made to the Georgia Department of Natural Resources, Environmental Protection Division.

Bases

The NRC is required under NEPA to maintain an awareness of environmental impacts causally related to the construction and operation of facilities licensed under its authority. Further, some of the ETS requirements are couched in terms of compliance with relevant permits, e.g., the NPDES permit, issued by other licensing authorities. The reports of exceeding limits of relevant permits also alert the NRC staff to environmental problems that may require mitigative action.

5.6.2 Nonroutine Reports

(Deleted. Refer to 10 CFR 50.72 and 10 CFR 50.73 for reporting requirements.)

5.6.3 Changes in Environmental Technical Specifications and Permits

5.6.3.1 Changes in Environmental Technical Specifications

Requests for changes in ETS shall be submitted to the NRC for review and authorization in accordance with 10 CFR 50.90. The request shall include an evaluation of the environmental impact of the proposed change and a supporting justification. Implementation of such requested changes in ETS shall not commence prior to incorporation by the NRC of the new specifications in the license.

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 86
License No. NPF-5

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 2 (the facility) Facility Operating License No. NPF-5 filed by Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, (the licensee) dated February 13, 1987, 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 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 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-5 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 86, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Kahtan N. Jabbour, Acting Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 1, 1987

L.C.

with change discussed

9/2/87 SB

no note

7/24/87

PD#II-3/DRP-I/II	PD#II-3/DRP-I/II	OGC-Bethesda	KNJ PD#II-3/DRP-I/II
MDuncan/rad	LCrocker	M. Karman	KJabbour Acting PD
06/ 1 /87	06/ /87	07/2 /87	12/01 /87

ATTACHMENT TO LICENSE AMENDMENT NO. 86

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. Corresponding overleaf pages are provided to maintain document completeness.

Remove Page	Insert Page
IIa	IIa
XV	XV
XVI	XVI
1-9	1-9
1-10	1-10
3/4 3-47	3/4 3-47
3/4 3-48	3/4 3-48
3/4 4-3	3/4 4-3
3/4 4-4	3/4 4-4
3/4 4-7	3/4 4-7
3/4 4-8	3/4 4-8
3/4 4-9	3/4 4-9
3/4 4-10	3/4 4-10
3/4 4-11	3/4 4-11
3/4 5-1	3/4 5-1
3/4 5-2	3/4 5-2
3/4 5-4	3/4 5-4
3/4 5-7	3/4 5-7
3/4 5-8	3/4 5-8
3/4 6-5	3/4 6-5
3/4 6-6	3/4 6-6
3/4 6-7	3/4 6-7
3/4 6-8	3/4 6-8
3/4 8-7	3/4 8-7
3/4 8-8	3/4 8-8
6-6	6-6
6-9	6-9
6-11	6-11
6-12	6-12
6-13	6-13
6-14	6-14
6-14a	6-14a
6-14b	6-14b
6-14d	6-14d
6-15	6-15
6-16	6-16
6-17	6-17

Replace the following pages of the Appendix B Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. Corresponding overleaf pages are provided to maintain document completeness.

4-1	4-1
4-2	4-2
5-8	5-8

INDEX

ADMINISTRATIVE CONTROLS

Correction letter of 8-2-86

ADMINISTRATIVE CONTROLS

VENTING

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

MILK ANIMAL

A cow or goat that is producing milk for human consumption.

REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATIONS

Notation	Definition	Frequency
S	Once per shift	Once per 12 hours
D	Daily	Once per 24 hours
W	Weekly	Once per 7 days
M	Monthly	Once per 31 days
Q	Quarterly	Once per 92 days
SA	Semi-annually	Once per 184 days
R	REFUELING	Once per 18 months
S/U	STARTUP	Prior to each reactor startup
P	Prior	Completed prior to each release
NA	Not applicable	Not applicable

Correction letter of 8-2-85

INSTRUMENTATION

SEISMIC MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6.2 The seismic monitoring instrumentation shown in Table 3.3.6.2-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.6.2.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.6.2-1.

4.3.6.2.2 Each of the above required seismic monitoring instruments actuated during a seismic event shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 30 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

TABLE 3.3.6.2-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs[a]		
a. Diesel Generator Building El 130'0"[c] (2L51-N021)	0-0.5g	1
b. Reactor Building 87' Level on Drywell Pedestal (2L51-N020)	0-0.5g	1
c. Drywell - Feedwater Inlet to RPV (2L51-N004)	0-0.5g	1
d. Switchyard[c] (1L51-N005)	0-0.5g	1
2. Triaxial Peak Recording Accelerometers		
a. Diesel Generator Base Support[c] (1L51-N007)	0-1.0g	1
b. Intake Structure[c] (1L51-N006)	0-1.0g	1
c. Control Building Main Control Room Floor[c] (1L51-N008)	0-1.0g	1
d. Control Building Floor El 112'[c] (2L51-N028)	0-1.0g	1
e. Reactor Bldg Refueling Floor (2L51-N029)	0-1.0g	1
f. Reactor Pedestal Inside Biological Shield (2L51-N035)	0-2.0g	1
g. Reactor Piping - Feedwater Inlet to RPV (2L51-N034)	0-2.0g	1
3. Triaxial Seismic Switches[b]		
a. Reactor Building 87' Level on Drywell Pedestal (2L51-N022)	0.025-0.25g	1
b. Reactor Building 185' Level Outside Biological Shield (2L51-N024)	0.025-0.25g	
4. Triaxial Response Spectrum Recorder[a]		
a. Hatch - Unit 1 Containment Foundation El 87'[c] (1L51-N105)	2-26 Hz 0-0.5g	1

[a] With main control room indication and annunciation.

[b] With main control room annunciation.

[c] Shared with Hatch - Unit 1.

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.3 An idle recirculation loop shall not be started unless the temperature differential between the reactor coolant within the dome and the bottom head drain is $\leq$ 145°F, and

a. The temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is $\leq$ 50°F when both loops have been idle, or

b. The temperature differential between the reactor coolant within the idle and operating recirculation loops is $\leq$ 50°F when only one loop has been idle, and the operating loop flow rate is $\leq$ 50% of rated loop flow.

APPLICABILITY: CONDITIONS 1, 2, 3 and 4.

ACTION:

With temperature differences and/or flow rate exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.3 The temperature differential and flow rate shall be determined to be within the limit within 30 minutes prior to startup of an idle recirculation loop.

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The safety valve function of the following reactor coolant system safety/relief valves shall be OPERABLE with the mechanical lift settings within ± 1% of the indicated pressures*.

4 Safety-relief valves @ 1090 psig.
4 Safety-relief valves @ 1100 psig**.
3 Safety-relief valves @ 1110 psig**.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

a. For low-low set valves, take the action required by Specification 3.4.2.2. For ADS valves, take the action required by Specification 3.5.2.
b. With one or more safety/relief valves stuck open, place the reactor mode switch in the Shutdown position.
c. With one or more S/RV tailpipe pressure switches of an S/RV declared inoperable and the associated S/RV(s) otherwise indicated to be open, place the reactor mode switch in the shutdown position.
d. With one S/RV tailpipe pressure switch of an S/RV declared inoperable and the associated S/RV(s) otherwise indicated to be closed, plant operation may continue. Remove the function of that pressure switch from the low low set logic circuitry until the next COLD SHUTDOWN. Upon COLD SHUTDOWN, restore the pressure switch(s) to OPERABLE status before STARTUP.
e. With both S/RV tailpipe pressure switches of an S/RV declared inoperable and the associated S/RV(s) otherwise indicated to be closed, restore at least one inoperable switch to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
f. The failure or malfunction of any safety/relief valve shall be reported by telephone within 24 hours; confirmed by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designee no later than the first working day following the event; and a written followup report within 30 days. The written followup report should be completed in accordance with 10 CFR 50.73 or other applicable requirements.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The tail-pipe pressure switches of each safety/relief valve shall be demonstrated OPERABLE by performance of:

a. CHANNEL FUNCTIONAL TEST:
   1. At least once per 31 days, except that all portions of the channel inside the primary containment may be excluded from the CHANNEL FUNCTIONAL TEST, and
   2. At each scheduled outage of greater than 72 hours during which entry is made into the primary containment, if not performed within the previous 31 days.
b. CHANNEL CALIBRATION and verifying the setpoint to be 85 psig, with an allowable tolerance of +15 psig and -5 psig, at least once per 18 months.

* The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperature and pressure.

** Up to two inoperable valves may be replaced with spare OPERABLE valves with lower setpoints of 1090 and 1100 psig, respectively, until the next refueling outage.

3/4.4.4 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.4 The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.4.4-1.

APPLICABILITY: At all times.

ACTION:

a. In CONDITION 1, 2 and 3:

1. With the conductivity or chloride concentration exceeding the limit specified in Table 3.4.4-1, but less than 10 µmho/cm at 25°C and less than 0.5 ppm, respectively, operation may continue for up to 24 hours and this need not be reported to the Commission, provided that operation under these conditions shall not exceed 336 hours per year. If operation under these conditions exceeds 336 hours per year, in lieu of any other report required by 10 CFR 50.73, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2, within 30 days, outlining the course of the limit violation and the plans for restoring the conductivity or chloride concentration to within the limit. The provisions of Specification 3.0.4 are not applicable.

2. With the conductivity or chloride concentration exceeding the limit specified in Table 3.4.4-1 for more than 24 hours during one continuous time interval or with the conductivity exceeding 10 µmho/cm at 25°C or chloride exceeding 0.5 ppm, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. At all other times:

1. With the conductivity of the reactor coolant in excess of the limit specified in Table 3.4.4-1, restore the conductivity to within the limit within 24 hours.

2. With the chloride limit of Table 3.4.4-1 exceeded for more than 48 hours, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system. Determine that the structural integrity of the reactor coolant system remains acceptable for continued operation prior to proceeding to CONDITION 3.

SURVEILLANCE REQUIREMENTS

4.4.4 The reactor coolant shall be determined to be within the specified chemistry limit by:

a. Analyzing a sample of the reactor coolant for conductivity and chlorides at least once per 72 hours, and

b. Continuously recording the conductivity of the reactor coolant, or

c. Analyzing a sample of the reactor coolant for conductivity at least once per 24 hours when the continuous recording conductivity monitor is inoperable.

TABLE 3.4.4-1

REACTOR COOLANT SYSTEM
CHEMISTRY LIMITS

OPERATIONAL CONDITION	CHLORIDES	CONDUCTIVITY (μmhos/cm @25°C)
1	< 0.5 ppm	< 5
2	< 0.1 ppm	< 5
At all other times	< 0.1 ppm	<10

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.5 The specific activity of the reactor coolant shall be limited to:

a. $\leq$ 0.2 μCi/gram DOSE EQUIVALENT I-131, and

b. $\leq 100/\bar{E}$ μCi/gram.

APPLICABILITY: CONDITIONS 1, 2, 3 and 4.

ACTION:

a. In CONDITIONS 1, 2 and 3, with the specific activity of the reactor coolant;

1. > 0.2 μCi/gram DOSE EQUIVALENT I-131 but $\leq$ 4.0 μCi/gram for more than 48 hours during one continuous time interval or > 4.0 μCi/gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

2. $> 100/\bar{E}$ μCi/gram, be in at least HOT SHUTDOWN with the main steamline isolation valves closed within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In CONDITION 1, 2, 3 or 4,

1. With the specific activity of the primary coolant > 0.2 μCi/gram DOSE EQUIVALENT I-131 or $> 100/\bar{E}$ μCi/gram, perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 at least once per 4 hours until the specific activity of the primary coolant is restored to within its limits.

ACTION: (Continued)

2. With:

   a) THERMAL POWER changed by more than 15% of RATED THERMAL POWER in one hour, or

   b) The off-gas level, at the SJAE, increased by more than 10,000 μ Ci/sec. in one hour at release rates less than 75,000 μ Ci/sec. or

   c) The off-gas level, at the SJAE, increased by more than 15% in one hour at release rates greater than 75,000 μ Ci/sec.,

   perform the sampling and analysis requirement of Item 4C of Table 4.4.5-1.

SURVEILLANCE REQUIREMENTS

4.4.5 The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.4.5-1.

3/4.5.1 HIGH PRESSURE COOLANT INJECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.1 The High Pressure Coolant Injection (HPCI) system shall be OPERABLE with:

a. One OPERABLE high pressure coolant injection pump, and

b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor pressure vessel.

APPLICABILITY: CONDITIONS 1*, 2* and 3* with reactor vessel steam dome pressure > 150 psig.

ACTION:

a. With the HPCI system inoperable, POWER OPERATION may continue and the provisions of 3.0.4 do not apply*, provided the RCIC system, ADS, CSS and LPCI system are OPERABLE; restore the inoperable HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ 150 psig within the following 24 hours.

b. With the surveillance requirements of Specification 4.5.1 not performed at the required frequencies due to low reactor steam pressure, the provisions of Specification 4.0.4 are not applicable provided the appropriate surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

SURVEILLANCE REQUIREMENTS

4.5.1 The HPCI shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water, and

*See Special Test Exception 3.10.5

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 92 days, by verifying that the system develops a flow of at least 4250 gpm for a system head corresponding to a reactor pressure of $\geq$ 1000 psig when steam is being supplied to the turbine at $\leq$ 1000 psig.

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

2. Verifying that the system develops a flow of at least 4250 gpm for a system head corresponding to a reactor pressure of $\geq$ 165 psig when steam is being supplied to the turbine at 165 $\pm$ 15 psig.

3. Verifying that the suction for the HPCI system is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank low water level signal and on a suppression chamber high water level signal.

3/4.5.3 LOW PRESSURE CORE COOLING SYSTEMS

CORE SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.3.1 Two independent Core Spray System (CSS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE CSS pump, and

b. An OPERABLE flow path capable of taking suction from at least one of the following OPERABLE sources and transferring the water through the spray sparger to the reactor vessel;

1. In CONDITION 1, 2 or 3, from the suppression pool.

2. In CONDITION 4 or 5*;

a) From the suppression pool, or

b) When the suppression pool is being drained, from the condensate storage tank containing at least 150,000 gallons of water.

APPLICABILITY: CONDITIONS 1, 2, 3, 4, and 5*.

ACTION

a. In CONDITION 1, 2 or 3;

1. With one CSS subsystem inoperable, POWER OPERATION may continue provided both LPCI subsystems are OPERABLE; restore the inoperable CSS subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both CSS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

* The core spray system and the suppression chamber are not required to be OPERABLE provided that the reactor vessel head is removed and the cavity is flooded, the spent fuel pool gates are removed, and the water level is maintained within the limits of Specification 3.9.9 and 3.9.10.

LOW PRESSURE COOLANT INJECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.3.2 Two independent Low Pressure Coolant Injection (LPCI) subsystems of the residual heat removal system (RHR) shall be OPERABLE with each subsystem comprised of:

a. Two OPERABLE RHR pumps,

b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor pressure vessel.

APPLICABILITY: CONDITIONS 1, 2, 3, 4* and 5*, **.

ACTION:

a. In CONDITION 1, 2 or 3;

1. With one LPCI subsystem or one LPCI pump inoperable, POWER OPERATION may continue provided both CSS subsystems are OPERABLE; restore the inoperable LPCI subsystem or pump to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both LPCI subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and either be in COLD SHUTDOWN or maintain reactor coolant temperature ≤ 400°F by use of alternate heat removal methods within the following 24 hours.

3. With the LPCI system cross-tie valve open or power not removed from the valve operator, be in at least HOT SHUTDOWN with 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In CONDITION 4* or 5*, ** with one or more LPCI subsystems inoperable, take the ACTION required by Specification 3.5.3.1. The provisions of Specification 3.0.3 are not applicable.

* Not applicable when two CSS subsystems are OPERABLE per Specification 3.5.3.1.

**Not applicable when the CSS is not required to be OPERABLE per Specification 3.5.3.1.

SURVEILLANCE REQUIREMENTS

4.5.3.2 Each LPCI subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water,

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

3. Verifying that the subsystem cross-tie valve is closed with power removed from the valve operator.

b. At least once per 92 days by verifying each pair of LPCI pumps discharging to a common header can be started from the control room and develops a total flow of at least 17,000 gpm against a system head corresponding to a reactor vessel pressure of $\geq$ 20 psig.

c. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet either .75 L_a, or .75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either .75 L_a or .75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either .75 L_a or .75 L_t, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a or 0.25 L_t,

2. Has a duration sufficient to establish accurately the change in leakage rate between the type A test and the supplemental test, and

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at P_a, 57.5 psig, or P_t, 28.8 psig.

d. Type B and C tests* shall be conducted at P_a, 57.5 psig, at intervals no greater than 24 months except for tests involving:

1. Air locks, which shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3, and

2. Main steam line isolation valves, which shall be leak tested at least once per 18 months.

e. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.

f. The provisions of Specification 4.0.2 are not applicable.

*All Type B and Type C Leakage Tests (i.e., Local Leak Rate Tests) that fail (i.e., test leakage is such that an LER would be required) during an outage shall be reported according to 10 CFR 50.73 by one 30-day written report that is due within 30 days of the first leakage test failure in the outage. All other leakage test failures discovered during the outage will be reported in a revision to the original report due within 30 days after the end of the outage.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT AIR LOCK

LIMITING CONDITION FOR OPERATION

3.6.1.3 The primary containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of $\leq 0.05\ L_a$ at P_a, 57.5 psig.

APPLICABILITY: CONDITIONS 1, 2* and 3.

ACTION:

a. With one primary containment air lock door inoperable, maintain at least the OPERABLE air lock door closed; restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed; operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 30 days. The provisions of Specification 3.0.4 are not applicable.

b. With the primary containment air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours.

c. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 The primary containment air lock shall be demonstrated OPERABLE:

a. **After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying no detectable seal leakage when the gap between the door seals is pressurized to ≥ 10 psig for at least 15 minutes.

b. At least once per 6 months by conducting an overall air lock leakage rate test at P_a, 57.5 psig, and by verifying that the overall air lock leakage rate is within its limit.

c. At least once per 6 months by verifying that only one door in the air lock can be opened at a time.

*See Special Test Exception 3.10.1.
**Exemption to Appendix J of 10 CFR 50.

MSIV LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.4 Two MSIV Leakage Control System (LCS) subsystems shall be OPERABLE.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

a. With one MSIV leakage control system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. The provisions of Specification 3.0.4 are not applicable.

b. With both MSIV leakage control system subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 Each MSIV Leakage Control System subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by starting the blower from the control room and operating the blower for at least 15 minutes.

b. Each COLD SHUTDOWN, if not performed within the previous 92 days, by cycling each bleeder valve through at least one complete cycle of full travel.

c. At least once per 18 months by performance of a functional test which includes simulated actuation of the subsystem throughout its operating sequence and verifying that each automatic valve actuates to its correct position and the blower starts and developes:

   1. For inboard MSIVs - 100 scfm at a vacuum of 60" H_2O, and

   2. For outboard MSIVs - 240 scfm at a vacuum of 50" H_2O.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The structural integrity of the primary containment shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of the accessible interior and exterior surfaces of the containment and verifying no apparent changes in appearance of the surfaces or other abnormal degradation.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.4 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to 10 CFR 50.73 or Specification 6.9.2, as applicable. If the number of failures in the last 100 valid tests, on a per nuclear unit basis, is ≥ 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.

TABLE 4.8.1.1.2-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures In Last 100 Valid Tests*	Test Frequency
≤ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
≥ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis.

MEETING FREQUENCY

6.5.1.4 The PRB shall meet at least once per calendar month and as convened by the PRB Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the PRB necessary for the performance of the PRB responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and three voting members including alternates.

RESPONSIBILITIES

6.5.1.6 The Plant Review Board shall be responsible for:

a. Review of all procedures required by Specification 6.8 and changes thereto, except those for the Radiological Environmental Monitoring Program, any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

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

c. Review of all proposed changes to Appendix "A" Technical Specifications.

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

e. Investigation of all reportable violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Vice President-Plant Hatch, the Senior Vice President-Nuclear Operations, and to the Safety Review Board (SRB).

f. Review of all REPORTABLE EVENTS.

g. Review of unit operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Plant Manager or the SRB.

QUORUM

6.5.2.6 The minimum quorum of the SRB necessary for the performance of the SRB review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least a majority of the members. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The SRB shall be responsible for the review of:

a. The safety evaluations for (1) changes to procedures, equipment or systems and (2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. All REPORTABLE EVENTS.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meetings minutes of the Plant Review Board.

k. The Radiological Environmental Monitoring Program and the results thereof annually.

l. The Offsite Dose Calculation Manual, Process Control Program, and implementing procedures at least once per 24 months.

AUTHORITY

6.5.2.9 The SRB shall report to and advise the Senior Vice President - Nuclear Operations on those areas of responsibility specified in Section 6.5.2.7 and 6.5.2.8.

RECORDS

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

a. Minutes of each SRB meeting shall be prepared, approved and forwarded to the Senior Vice President-Nuclear Operations within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Senior Vice President-Nuclear Operations within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Senior Executive Vice President, the Senior Vice President-Nuclear Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and/or 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 PRB, and the results of this review shall be submitted to the SRB, the Vice President-Plant Hatch, and the Senior Vice President-Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

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

a. The unit shall be placed in at least HOT SHUTDOWN within two hours.

b. The Safety Limit violation shall be reported to the Commission as soon as practical and in all cases within one hour of occurrence. The Vice President-Plant Hatch, the Senior Vice President-Nuclear Operations and the SRB shall be notified within 24 hours.

SAFETY LIMIT VIOLATION (Continued)

c. A Licensee Event Report shall be prepared pursuant to 10 CFR 50.73.

d. The Licensee Event Report shall be submitted to the Commission in accordance with 10 CFR 50.73, and to the PRB, the SRB, the Vice President-Plant Hatch, and the Senior Vice President-Nuclear Operations within 30 days of the violation.

6.8 PROCEDURES

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

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

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

6.8.2 Each procedure of 6.8.1 and other procedures which the Plant Manager or Plant Support Manager has determined to affect nuclear safety, and changes thereto, shall be reviewed by the PRB and approved by the appropriate member of plant management, designated by the Plant Manager or Plant Support Manager, prior to implementation. The Plant Manager or Plant Support Manager will approve administrative procedures, security plan implementing procedures, and changes thereto. The Manager-Plant Training and Onsite Emergency Preparedness shall approve the emergency plan implementing procedures and changes thereto. All other procedures of this specification and changes thereto will be approved by the department head designated by the Plant Manager or Plant Support Manager. The procedures of this specification shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

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 the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

START-UP REPORT

6.9.1.1 A summary report of plant start-up 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 plant.

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

6.9.1.3 Start-up reports shall be submitted within (1) 90 days following completion of the start-up test program, or (2) 90 days following resumption or commencement of commercial power operation, or (3) 12 months following initial criticality, whichever is earliest. If the Start-up Report does not cover all three events (i.e., initial criticality, completion of start-up test program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1]

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

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

ANNUAL REPORTS (Continued)

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility and other personnel, including contractors, receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,[2] e.g., reactor operations and surveillance inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

b. Documentation of all challenges to safety/relief valves.

c. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

d. Any other unit unique reports required on an annual basis.

ANNUAL RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE REPORT[a]

6.9.1.6 Routine radiological environmental surveillance reports covering the radiological environmental surveillance activities related to the plant during the previous calendar year shall be submitted prior to May 1 of each year. A single report may fulfill this requirement for both units.

6.9.1.7 The Annual Radiological Environmental Surveillance Report shall include summaries, interpretations, and statistical evaluation of the

---

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

[2]This tabulation supplements the requirements of 20.407 of 10 CFR Part 20.

results of the radiological environmental surveillance activities for the reporting period, including (as appropriate) a comparison with the preoperational studies, operational controls, previous environmental surveillance reports, and an assessment of any observed impacts of the plant operation on the environment. The reports shall also include the results of the land use surveys required by Specification 3.16.2 of Unit 1 Technical Specifications and the results of licensee participation in the interlaboratory comparison program required by Specification 3.16.3 of Unit 1 Technical Specifications.

The Annual Radiological Environmental Surveillance Report shall include summarized and tabulated results in the format of table 6.9.1.7-1 of all radiological environmental samples taken during the report period, with the exception of naturally occurring radionuclides which need not be reported. In the event that some results are not available for inclusion with the report, the report shall be submitted, noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as practicable in a supplementary report.

The reports shall also include the following:

a. Summary description of the radiological environmental monitoring program.

b. Map of all sampling locations as keyed to a table indicating distances and directions from main stack.

c. Results of the licensee participation in the Interlaboratory Comparison Program.

SEMI-ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT[a]

6.9.1.8 Routine radioactive effluent release reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

Any changes to the ODCM shall be submitted with the next semi-annual report in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the monthly operating report for the period in which the evaluation was reviewed and accepted by the Plant Review Board.

a. A single submittal may be made for a multiple-unit station. The submittal should combine those sections that are common to all units at the station; however, the submittal shall specify the releases of radioactive material from each unit.

TABLE 6.9.1.7-1

ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM SUMMARY

Name of Facility Edwin I. Hatch Nuclear Plant Docket No. 50-321, 50-366

Location of Facility Appling County, Georgia Reporting Period ________

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection[a]	All Indicator Locations Mean Range	Location of Highest Annual Mean: Name Distance, and Direction	Location of Highest Annual Mean: Mean Range[b]	Control Locations Mean Range	Number of REPORTABLE EVENTS

a. Lower Limit of Detection is defined in table notation a. of table 4.16.1-1, Specification 4.16.1 of Unit 1.

b. Mean and range based upon detectable measurements only. Fraction of detectable measurements at specified locations is indicated in parentheses.

e. Type of container, e.g., LSA, type A, type B, large quantity

f. Solidification agent, e.g., cement.

The Radioactive Effluent Release Report shall include (on a quarterly basis) unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents that were in excess of 1 Ci, excluding dissolved and entrained gases and tritium for liquid effluents, or those in excess of 150 Ci of noble gases or 0.02 Ci of radioiodines for gaseous releases.

The Radioactive Effluent Release Report shall include any changes to the PROCESS CONTROL PROGRAM and to the OFFSITE DOSE CALCULATION MANUAL made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the Regional Office of Inspection and Enforcement no later than the 15th of each month following the calendar month covered by the report.

(This page is intentionally left blank.)

(This page is intentionally left blank.)

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. Special reports for fire protection equipment operating and surveillance requirements shall be submitted, as required, by the Fire Hazards Analysis and its Appendix B requirements.

6.10 RECORD RETENTION

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.

6.10.1 The following records shall be retained for at least five 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.

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.

4.0 Special Surveillance and Study Activities

4.1 Erosion Control Inspection

4.2 Unusual or Important Events Requirements

Requirements

The licensee shall be alert to the occurrence of unusual or important events. Unusual or important events are those that cause potentially significant environmental impact or could be of public interest concerning environmental impact from plant operation. The following are examples: unusual or important bird impaction events on cooling tower structures or meteorological towers, onsite plant or animal disease outbreaks, unusual mortality of any species protected by the Endangered Species Act of 1973, fish kills near the HNP site, and significant violations of relevant permits and certifications.

Actions

Should an unusual or important event occur, the licensee shall make a prompt report to the NRC as required by 10 CFR 50.72 or 10 CFR 50.73.

Bases

Prompt reporting to the NRC of unusual or important events, as described, is necessary for responsible and orderly regulation of the nation's system of nuclear power reactors. The information thus provided may be useful or necessary to others concerned with the same environmental resources. Prompt knowledge and action may serve to alleviate the magnitude of environmental impact or to place it into a perspective broader than that available to the licensee. The NRC also has an obligation to be responsive to inquiries from the public and the news media concerning potentially significant environmental events at nuclear power stations.

4.3 Exceeding Limits of Other Relevant Permits

Requirements

The licensee shall notify the NRC of occurrences exceeding the limits specified in relevant permits and certificates issued by other Federal, State, and local agencies that are reportable to the agency that issued the permit. This requirement shall apply only to topics of NEPA concern within the NRC area of responsibility as identified in the Environmental Technical Specifications (ETS).

This requirement shall commence with the date of issuance of the operating license for Unit 2 and continue until approval for modification or termination is obtained from the NRC in accordance with section 5.6.3.

Action

The licensee shall make a report to the NRC in the event of a REPORTABLE EVENT exceeding a limit specified in a relevant permit or certificate issued by another Federal, State, or local agency. The report shall be submitted within the time limit specified by the reporting requirements of the corresponding certification or permit issued pursuant to Section 401 or 402 of PL 92-500. The report will consist of a copy of the report made to the Georgia Department of Natural Resources, Environmental Protection Division.

Bases

The NRC is required under NEPA to maintain an awareness of environmental impacts causally related to the construction and operation of facilities licensed under its authority. Further, some of the ETS requirements are couched in terms of compliance with relevant permits, e.g., the NPDES permit, issued by other licensing authorities. The reports of exceeding limits of relevant permits also alert the NRC staff to environmental problems that may require mitigative action.

5.6.2 Nonroutine Reports

Deleted. Refer to 10 CFR 50.72 and 10 CFR 50.73 for reporting requirements.

5.6.3 Changes in Environmental Technical Specifications and Permits

5.6.3.1 Changes in Environmental Technical Specifications

Requests for changes in ETS shall be submitted to the NRC for review and authorization in accordance with 10 CFR 50.90. The request shall include an evaluation of the environmental impact of the proposed change and a supporting justification. Implementation of such requested changes in ETS shall not commence prior to incorporation by the NRC of the new specifications in the license.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS.149 AND 86 TO

FACILITY OPERATING LICENSES DPR-57 AND NPF-5

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

I. INTRODUCTION

By submittal dated February 13, 1987 (Reference 1) Georgia Power Company (GPC, the licensee) proposed changes to the Technical Specifications (TS) for the Edwin I. Hatch Nuclear Plant, Units 1 and 2, to revise the reporting requirements to conform to Sections 50.72 and 50.73 of 10 CFR Part 50; and to revise the reporting requirements regarding the iodine activity level in the primary coolant and delete the requirement for plant shutdown in the event iodine activity level in the primary coolant exceeds the limits for more than 800 hours in a 12-month period. The proposed changes to the reporting requirements to bring the TS into conformance with 10 CFR Part 50, Sections 50.72 and 50.73 are in response to NRC Generic Letter 83-43 (Reference 2), while the proposed changes regarding reporting and plant shutdown as a result of high iodine activity levels are in response to NRC Generic Letter 85-19 (Reference 3).

II. EVALUATION

The proposed changes are divided into 12 parts as discussed and evaluated below:

2.1 Proposed Change 1:

This proposed change would add to the Unit 1 Technical Specifications Definitions Section the term REPORTABLE EVENT. This proposed change would also delete from the Unit 2 Technical Specifications Definitions section the term REPORTABLE OCCURRENCE and replace it with the term REPORTABLE EVENT, as follows:

> A "REPORTABLE EVENT" shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

This change would appear in Unit 1 Technical Specifications Section 1.0, item III. The change would also appear in Unit 2 Technical Specifications Section 1.0

These changes are administrative in nature, changing the definitions appearing in the TS of the two units to conform to the wording presented in Generic Letter 83-43. They are, therefore, acceptable.

8712220168 871201
PDR ADOCK 05000321
P PDR

2.2 Proposed Change 2:

This proposed change would relocate the reporting requirements for the failure or malfunction of any safety/relief valve from Section 6.9.1.12 of the Units 1 and 2 Technical Specifications to Section 3.6.H.1.a of the Unit 1 Technical Specifications and section 3.4.2.1.f of Unit 2 Technical Specifications. The written followup report requirement would be changed from 14 days to 30 days.

This reporting requirement is being relocated rather than deleted, because certain types of safety/relief valve malfunctions (e.g., the opening of a relief valve at a pressure lower than the setpoint) do not fall under the provisions of 10 CFR 50.72(b)(1)(ii) or 10 CFR Part 50.73(a)(2)(ii). These types of malfunctions should still be reported in accordance with Item II.K.3.3 of NUREG 0737. Failures or malfunctions of safety/relief valves, which do not fall under the provisions of 10 CFR 50.72 and 10 CFR 50.73, would still be subject to the reporting requirements contained herein. The change in the written followup report requirement from 14 to 30 days is made to be consistent with the requirements of 10 CFR 50.73.

These proposed changes modify the reporting requirements to be consistent with 10 CFR 50.72 and 10 CFR 50.73, while still retaining the requirement for near-term reporting of safety/relief valve malfunctions that do not fall under the provisions of 10 CFR 50.72 and 10 CFR 50.73. They are, therefore, acceptable.

2.3 Proposed Change 3:

This proposed change would delete from the Unit 2 Technical Specifications the Special Report requirements for: (1) High Pressure Coolant Injection (HPCI) system actuation contained in Section 3.5.1.c, (2) Core Spray System (CSS) actuation contained in Section 3.5.3.1.a.3, and (3) Low Pressure Coolant Injection (LPCI) system actuation contained in Section 3.5.3.2.a.4.

These Special Reports are no longer required and may be deleted, because any actuation of these Emergency Core Cooling Systems (ECCS) is required to be reported under the:

1. Immediate notifications provisions of 10 CFR 50.72(b)(1)(iv) for "Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal."

2. Immediate notification provisions of 10 CFR 50.72(b)(2)(ii) for "Any event or condition that results in manual or automatic actuation of an Engineered Safety Feature (ESF)." Each system identified above as part of the ECCS is also an ESF system.

Also, written notification within 30 days is required by 10 CFR 50.73 (a)(2)(iv), which states that the licensee shall report "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF)." 10 CFR 50.73(g) states: "The requirements contained in this section replace all existing requirements for licensees to report "Reportable Occurrences" as defined in individual plant Technical Specifications." Deletion of these Special Reports from the TS is in accordance with the requirements of 10 CFR 50.73 and, therefore, is acceptable.

2.4 <u>Proposed Change 4:</u>

This proposed change would relocate the reporting requirements for Type B and Type C Leakage Tests from Unit 1 and Unit 2 Technical Specifications Sections 6.9.1.13 (which are proposed for deletion) to Sections 4.7.A.2.f and 4.7.A.2.g of the Unit 1 Technical Specifications and to Section 4.6.1.2.d of the Unit 2 Technical Specifications. The reporting requirements will continue to reflect the fact that one 30-day written report may be submitted within 30 days of the first leakage test failure which occurs during the outage, and that all other leakage test failures discovered during the outage will be reported in a revision to the original report, due within 30 days following the completion of the outage.

This relocation of reporting requirements is consistent with the requirements of 10 CFR 50.73, is clarifying in nature, and does not represent any change to the design, operation, or safety of the plant. It is, therefore, acceptable.

2.5 <u>Proposed Change 5</u>

Existing TS 3.3.6.2.a for Unit 2 requires a special 10-day report to be submitted in the event any of the required seismic monitoring instrumentation is inoperable for a period of more than 30 days. Existing TS 4.3.6.2.2 for Unit 2 requires a special report to be submitted within 10 days describing the magnitude, frequency spectrum and resultant effect of any seismic event on the features of the plant that are important to safety. In each case, a clause in these existing TS states that these reports are to be submitted "in lieu of any other report required by Specification 6.9.1." Generic Letter 83-43 stated that: "Some Technical Specifications currently require Special Reports or other routine reporting of events in lieu of a Licensee Event Report. Such reports are still required but the technical specification wording will need revision. These changes, where applicable, should also be included in your amendment request to modify your technical specifications."

TS 6.9.1.11, "Reportable Occurrences", in the existing TS is to be deleted by this amendment. Thus, consistent with the instructions in the generic letter, the licensee proposes to substitute "10 CFR 50.73" for the words "Specification 6.9.1" in the clauses of TS 3.3.6.2.a and TS 4.3.6.2.2. However, seismic monitoring instrumentation is not subject to the reporting requirements of 10 CFR 50.73. Therefore, rather than make the wording substitution requested by the licensee, the NRC staff has elected simply to delete the clauses. The resultant wording still requires the special 10-day reports as required by the existing TS and thus represents no change to the TS requirements. The changes are, therefore, acceptable.

This change to the licensee's request has been discussed with and is acceptable to the licensee. It does not represent a substantive change from the licensee's original request (Reference 1) as noticed in the Federal Register on July 15, 1987.

2.6 Proposed Change 6:

This proposed change would revise the wording of TS Section 4.8.1.1.4 to require that all diesel generator failures be reported pursuant to the new 10 CFR 50.73 rather than pursuant to the existing Specification 6.9.1. The change also provides for submission of the reports to the Regional office as well as to NRC headquarters. This change is in accordance with the instructions of Generic Letter 83-43 and is, therefore, acceptable.

2.7 Proposed Change 7:

This proposed change would delete Section 3.11.D and Bases 3.11.D from the Unit 1 Technical Specifications. Section 3.11.D presently provides that a Reportable Occurrence report shall be submitted for any event in which any of the limiting values of the Average Planar Linear Heat Generation Rate (APLHGR), the Linear Heat Generation Rate (LHGR), or the Minimum Critical Power Ratio (MCPR) are exceeded.

In accordance with 10 CFR 50.73(g), it is no longer necessary to retain a specific section in the Unit 1 Technical Specifications delineating the reporting requirements for this type of event. If a limiting value of APLHGR, LHGR, or MCPR is exceeded, and the required Technical Specification Action statement is not met, reporting under 10 CFR 50.73 (a)(2)(i)(B) is appropriate. This change, therefore, is acceptable.

2.8 Proposed Change 8:

This proposed change would revise the Unusual or Important Events reporting requirements listed in Section 4.2 of the Unit 1 and 2 Environmental Technical Specifications (ETS) so that reports are made as required by 10 CFR 50.72 and 10 CFR 50.73 rather than in accordance with ETS Section 5.6.2 which would be deleted. Also, the proposed change would reword the reporting requirements stated in Section 4.3 of the Unit 1 and 2 ETS so that in the case of a reportable event in which a limit specified in a relevant permit or certificate issued by another Federal, State, or local agency is exceeded, the report would be submitted within the time limit specified by the reporting requirements of the corresponding certification or permit issued pursuant to Section 401 or 402 of PL 92-500. The report which would be sent to the NRC would consist of a copy of the report which would be made to the Georgia Department of Natural Resources, Environmental Protection Division. These reportable event reporting requirements are currently contained in

Section 5.6.2. In addition, the proposed change would delete Sections 5.6.2 of the Unit 1 and 2 ETS which contain the reporting requirements for non-routine reports.

Reporting requirements for unusual or important environmental events are covered under the provisions of 10 CFR 50.72(b)(2)(vi), 10 CFR 50.73 (a)(2)(iii) and 10 CFR 50.73(a)(2)(x). Therefore, the reference to the reporting requirements presented in 10 CFR 50.72 and 10 CFR 50.73 is appropriate. The rewording of the reporting requirements stated in Section 4.3 does not constitute a change in the reporting requirements, but is merely a relocation of those requirements from Section 5.6.2 to Section 4.3, and a replacement of the term REPORTABLE OCCURRENCE with the term REPORTABLE EVENT. This change is consistent with 10 CFR 50.73 and is, therefore, acceptable.

2.9 Proposed Change 9:

This change would modify Section 3.4.4.a.1 of the Unit 2 Technical Specifications by deleting the reference to Section 6.9.1.9 reporting requirements and adding a reference to the Special Report requirements of Section 6.9.2 of the TS. This change is consistent with the instructions of Generic Letter 83-43 and is acceptable.

2.10 Proposed Change 10:

This proposed change would delete from Section 4.6.1.5 of the Unit 2 Technical Specifications the sentence: "Any abnormal degradation of the primary containment detected during the required inspections shall be reported to the Commission pursuant to Specification 6.9.1." The "required inspections" occur during shutdown.

Basis for Proposed Change:

This reporting requirement would be deleted because any event such as this, found while the reactor is shut down, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant, (including its principal safety barriers) being seriously degraded, requires a Four-Hour Report under the provisions of 10 CFR 50.72(b)(2)(i). In addition, any event or conditions such as this which resulted in the condition of the nuclear power plant, including its principal safety barriers being seriously degraded would require a Licensee Event Report written notification within 30 days under the provisions of 10 CFR 50.73 (a)(2)(ii). Deletion of this sentence is consistent with 10 CFR 50.73(g) and is, therefore, acceptable.

2.11 Proposed Change 11:

This proposed change would revise the ADMINISTRATIVE CONTROLS sections and associated TABLE OF CONTENTS and INDEX to reflect the revised immediate notification requirements of 10 CFR 50.72 and the Licensee Event Report system requirements of 10 CFR 50.73. Unit 1 Technical Specifications Sections 6.5.1.6.f, 6.5.2.7.g, 6.6, 6.6.1.a, 6.6.1.b, 6.7.1.b, 6.7.1.c, 6.7.1.d, 6.9, 6.10.1.c, and Table 6.9.1.7-1 would be appropriately reworded; and Sections 6.9.1.11, 6.9.1.12 and 6.9.1.13 would be deleted. Unit 2 Technical Specifications Section 6.9 of the Index, Sections 6.5.1.6.f, 6.5.2.7.g, 6.6, 6.6.1.a, 6.6.1.b, 6.7.1.b, 6.7.1.c, 6.7.1.d, 6.9, 6.10.1.c, and Table 6.9.1.7-1 would be appropriately reworded; and sections 6.9.1.11, 6.9.1.12, and 6.9.1.13 would be deleted.

These changes would make modifications of an administrative nature to the Unit 1 and 2 Technical Specifications, as specifically directed by Generic Letter 84-43, in order to incorporate the current reporting requirements stated in 10 CFR Part 50, Sections 50.72 and 50.73. These changes are consistent with the model technical specification presented in Standard Technical Specifications format in NRC Generic Letter 83-43, and are in accordance with 10 CFR 50.73(g). They are, therefore, acceptable.

2.12 <u>Proposed Change 12:</u>

This proposed change would make the following revisions to the Technical Specifications pursuant to Generic Letter 85-19, "Reporting Requirements on Primary Coolant Iodine Spike,":

In the Unit 1 Technical Specifications:

1. Delete the second paragraph of Section 3.6.F.1 which specifies iodine activity limits within a 12-month period.

2. Change the third paragraph of Section 3.6.F.1 to redefine the activity concentration limits.

3. Change the last paragraph of Section 4.6.F.1 to remove the 30-day reporting requirement.

4. Add paragraph 6.9.1.5.c (relocate existing paragraph 6.9.1.5.c to 6.9.1.5.d) to include requirements for an Annual Report.

In the Unit 2 Technical Specifications:

1. Delete paragraph 3.4.5.a.1 which limits iodine activity limits in a 12-month period.

2. Renumber paragraph 3.4.5.a.2 to 3.4.5.a.1 and redefine the activity concentration limits.

3. Renumber paragraph 3.4.5.a.3 to 3.4.5.a.2.

4. Change paragraph 3.4.5.b.1 to remove the REPORTABLE OCCURRENCE report requirement.

5. Change paragraph 3.4.5.b.2 to remove the 92-day Special Report requirement.

6. Add paragraph 6.9.1.5.c (relocate existing paragraph 6.9.1.5.c to 6.9.1.5.d) to include requirements for an Annual Report.

This change would make modifications to the Technical Specifications, as directed by Generic Letter 85-19. The reporting requirements for iodine spiking contained in the Technical Specifications would be changed from a short-term report to an item for inclusion in the Annual report. The information to be included in the Annual Report is similar to that previously required but has changed to more clearly designate the results to be included from the specific activity analysis and to delete the information regarding fuel burnup by core region. Also, the existing requirements contained in the Technical Specifications for plant shutdown, if coolant iodine activity limits are exceeded for 800 hours in a 12-month period, would be removed.

These changes are justified because, as discussed in Generic Letter 85-19, the quality of nuclear fuel has been greatly improved over the past decade with the result that normal coolant iodine activity (i.e., in the absence of iodine spiking) is well below the limit. Appropriate actions would be initiated long before accumulating 800 hours above the iodine activity limit. In addition, 10 CFR 50.72 (b)(1)(ii) requires the NRC to be immediately notified of fuel cladding failures that exceed expected values or that are caused by unexpected factors. Therefore, this Technical Specification limit is no longer necessary on the basis that proper fuel management and existing reporting requirements should preclude ever approaching the limit. Plant Hatch would continue to monitor iodine activity in the primary coolant and take responsible actions to maintain it at a reasonably low level (i.e., accumulated time with high iodine activity would not approach 800 hours). These changes are, therefore, acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

The amendments involve a change in use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in reporting requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there should be no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9). Pursuant to 10 CFR §51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (52 FR 26587) on July 15, 1987, and consulted with the state of Georgia. No public comments were received, and the state of Georgia did not have any comments.

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCES

1. Letter from James P. O'Reilly, Georgia Power Company, to the NRC, dated February 13, 1987.

2. Letter from Darrel G. Eisenhut, NRC, to all licensees and applicants for operating power reactors and holders of construction permits for operating reactors, "Reporting Requirements of 10 CFR Part 50, Sections 50.72 and 50.73, and Standard Technical Specifications (Generic Letter 83-43), dated December 19, 1983.

3. Letter from Hugh L.Thompson, NRC, to all licensees and applicants for operating power reactors and holders of construction permits for power reactors, "Reporting Requirements on Primary Coolant Iodine Spikes (Generic Letter 85-19), September 27, 1985.

Principal Contributor: L. Crocker

Dated: December 1, 1987