



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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 21
License No. NPF-37

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Commonwealth Edison Company (the licensee) dated January 18, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specification as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-37 is hereby amended to read as follows:

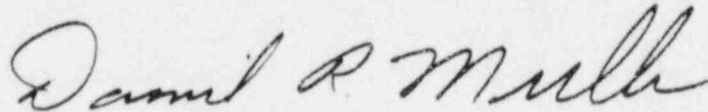
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(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 2i and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION



Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 27, 1988



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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 21
License No. NPF-66

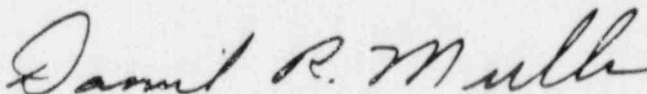
1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Commonwealth Edison Company (the licensee) dated January 18, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-66 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 21 and revised by Attachment 2 to NPF-60, and the Environmental Protection Plan contained in Appendix B, both of which are attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. Attachment 2 contains a revision to Appendix A which is hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION



Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 27, 1988

ATTACHMENT TO LICENSE AMENDMENT NOS. 21 AND 21
FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66
DOCKET NOS. STN-50-454 AND STN 50-455

Revise Appendix A as follows:

<u>Remove Pages</u>	<u>Insert Pages</u>
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
3/4 4-40	3/4 4-40
3/4 5-1	3/4 5-1
3/4 6-23	3/4 6-23
3/4 7-14	3/4 7-14
B3/4 4-5	B3/4 4-5
B3/4 4-6	B3/4 4-6
5-6	5-6
6-7	6-7
6-8	6-8
6-13	6-13
6-18	6-18
	6-18a

REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

- a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and
- b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

- a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and
- b. With the specific activity of the reactor coolant greater than $100/\bar{E}$ microCuries per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

* With T_{avg} greater than or equal to 500°F.

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

PS_{MAX} - MAXIMUM ALLOWABLE PORV SETPOINT (PSIG)

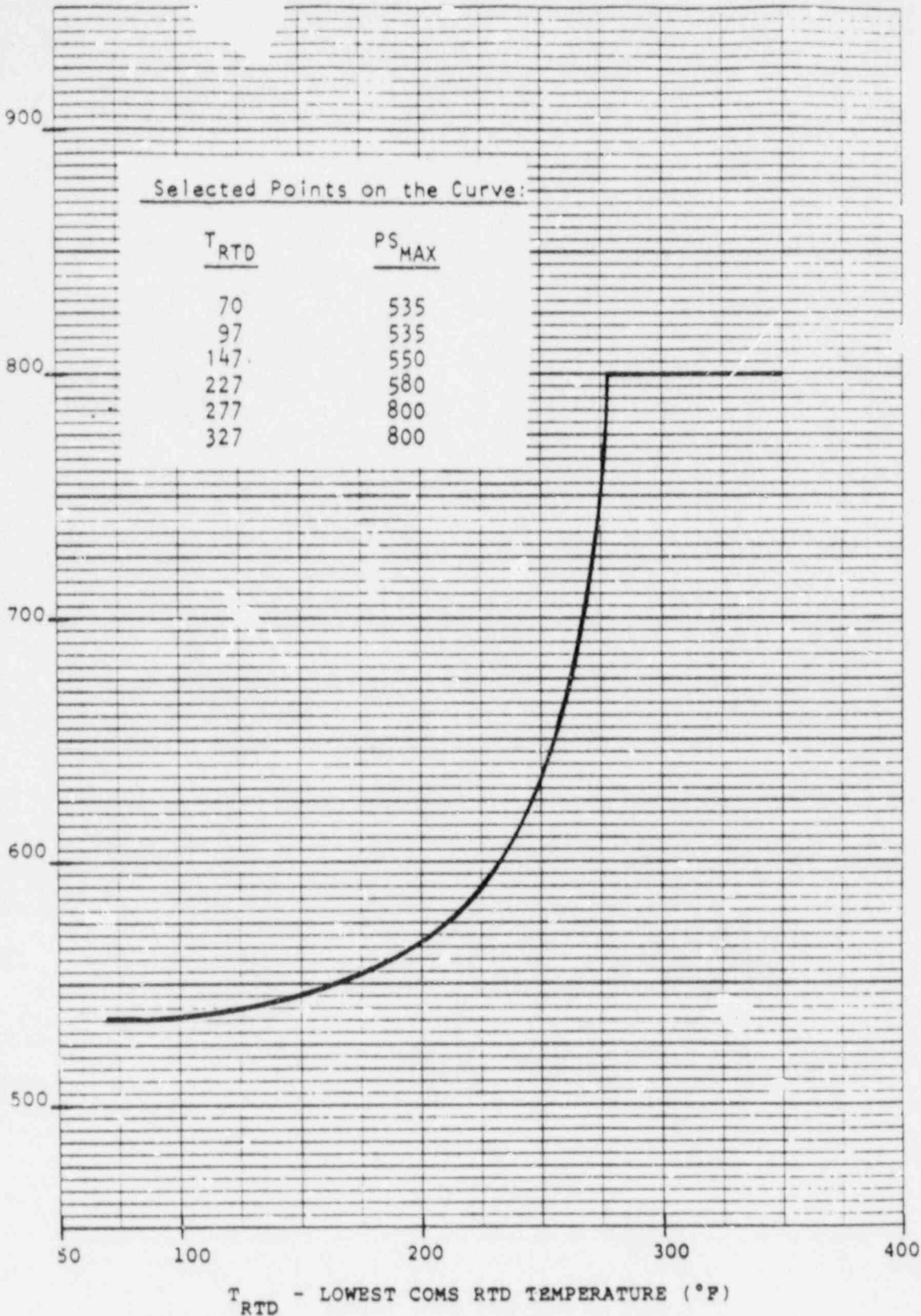


FIGURE 3.4-4
NOMINAL PORV PRESSURE RELIEF SETPOINT VERSUS
RCS TEMPERATURE FOR THE COLD OVERPRESSURE PROTECTION SYSTEM
APPLICABLE UP TO 10 EFPY

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

- a. The isolation valve open and power removed,
- b. A contained borated water level of between 31% and 63%,
- c. A boron concentration of between 1900 and 2100 ppm, and
- d. A nitrogen cover-pressure of between 602 and 647 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

- a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

- a. At least once per 12 hours by:
 - 1) Verifying the contained borated water level and nitrogen cover-pressure in the tanks, and
 - 2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

<u>PENETRATION</u>	<u>VALVE NO.</u>	<u>FUNCTION</u>	<u>MAXIMUM ISOLATION TIME (SEC)</u>
9. <u>Manual</u> (Continued)			
99	FW0150*,#	Feedwater	N.A.
100	FW015A*,#	Feedwater	N.A.
101	FW015B*,#	Feedwater	N.A.
102	FW015C*,#	Feedwater	N.A.
10. <u>Check</u>			
28	CV8113	RCP Seal Water Return	N.A.
37	CV8348*	RCS Loop Fill	N.A.
6	W007A	Chilled Water	N.A.
10	W007B	Chilled Water	N.A.
21	CC9534	RCP Mtr Brng Return	N.A.
24	CC9518	RCP Thermal Barrier Return	N.A.
25	CC9486	RCP Cooling Wtr Supply	N.A.
1	CS008A	Containment Spray	N.A.
16	CS008B	Containment Spray	N.A.
39	IA091	Instrument Air	N.A.
30	WM191	Make-Up Demin	N.A.
52	PR032	Process Radiation	N.A.
AL	PR002G	Process Radiation	N.A.
AL	PR002H	Process Radiation	N.A.
12	PS231A	Hydrogen Monitor	N.A.
31	PS231B	Hydrogen Monitor	N.A.
27	RY8047	PRT Nitrogen	N.A.
44	RY8046	PRT Make-Up	N.A.
26	SI8815*	Safety Injection	N.A.
50	SI8818A*	Safety Injection	N.A.
50	SI8818D*	Safety Injection	N.A.
51	SI8818B*	Safety Injection	N.A.
51	SI8818C*	Safety Injection	N.A.
59	SI8905A*	Safety Injection	N.A.
59	SI8905D*	Safety Injection	N.A.
60	SI8819A*	Safety Injection	N.A.
60	SI8819B*	Safety Injection	N.A.

PLANT SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

- c. With one essential service water makeup pump inoperable, restore the essential service water makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- d. With the essential service water pump discharge water temperature not meeting the above requirement, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- e. With the minimum Rock River water level not meeting the above requirement, notify the NRC within 1 hour in accordance with the procedure of 10 CFR 50.72 of actions or contingencies to ensure an adequate supply of cooling water to the Byron Station for a minimum of 30 days, verify the Rock River flow within 1 hour, and:
 - (1) If Rock River flow is less than 700 cubic feet per second (cfs) be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours, or
 - (2) If Rock River flow is equal to or greater than 700 cfs continue verification procedure every 12 hours or until Rock River water level exceeds 670.6 feet MSL or
 - (3) If Rock River level is equal to or less than 664.7 feet MSL be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours
- f. With one deep well inoperable and:
 - (1) The Rock River water level predicted, through National Weather Service flood forecasts, to exceed 702 feet MSL, or
 - (2) The Rock River water level at or below 670.6 feet MSL, or
 - (3) A tornado watch issued by the NWS that includes the area for the Byron Station.

Notify the NRC within 1 hour in accordance with the procedure of 10 CFR 50.72 of actions or contingencies to ensure an adequate supply of cooling water to the Byron Station for a minimum of 30 days and restore both wells to OPERABLE status before the Rock River water level exceeds 702 feet MSL or the minimum Rock River level or flow falls below 664.7 feet MSL or 700 cfs, respectively, or within 72 hours, whichever occurs first, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The UHS shall be determined OPERABLE at least once per:

- a. 24 hours by verifying the water level in each UHS cooling tower basin to be greater than or equal to 873.75 feet MSL. (50%),

REACTOR COOLANT SYSTEM

BASES

OPERATIONAL LEAKAGE (Continued)

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Byron Station, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

REACTOR COOLANT SYSTEM

BASES

SPECIFIC ACTIVITY (Continued)

The sample analysis for determining the gross specific activity and \bar{E} can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is related to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinct window for determination of the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample with typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

<u>COMPONENT</u>	<u>CYCLIC OR TRANSIENT LIMIT</u>	<u>DESIGN CYCLE OR TRANSIENT</u>
Reactor Coolant System	200 heatup cycles at $\leq 100^{\circ}\text{F}/\text{h}$ and 200 cooldown cycles at $< 100^{\circ}\text{F}/\text{h}$.	Heatup cycle - T_{avg} from $\leq 200^{\circ}\text{F}$ to $> 550^{\circ}\text{F}$. Cooldown cycle - T_{avg} from $> 550^{\circ}\text{F}$ to $\leq 200^{\circ}\text{F}$.
	200 pressurizer cooldown cycles at $\leq 200^{\circ}\text{F}/\text{h}$.	Pressurizer cooldown cycle temperatures from $> 650^{\circ}\text{F}$ to $\leq 100^{\circ}\text{F}$.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	$> 15\%$ of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential $> 320^{\circ}\text{F}$.
	200 leak tests.	Pressurized to ≥ 2485 psig.
	10 hydrostatic pressure tests.	Pressurized to ≥ 3107 psig.
	Secondary Coolant System	1 large steam line break.
10 hydrostatic pressure tests.		Pressurized to ≥ 1481 psig.

6.5 REVIEW INVESTIGATION AND AUDIT (Continued)

OFFSITE

6.5.1 The Superintendent of the Offsite Review and Investigative Function shall be appointed by the Manager of Nuclear Safety responsible for nuclear activities. The audit function shall be the responsibility of the Manager of Quality Assurance and shall be independent of operations.

a. Offsite Review and Investigative Function

The Superintendent of the Offsite Review and Investigative Function shall: (1) provide directions for the review and investigative function and appoint a senior participant to provide appropriate direction, (2) select each participant for this function, (3) select a complement of more than one participant who collectively possess background and qualifications in the subject matter under review to provide comprehensive interdisciplinary review coverage under this function, (4) independently review and approve the findings and recommendations developed by personnel performing the review and investigative function, (5) approve and report in a timely manner all findings of non-compliance with NRC requirements to the Station Manager, Assistant Vice President and General Manager - Nuclear Stations, Manager of Quality Assurance, and the Vice President - Nuclear Operations. During periods when the Superintendent of Offsite Review and Investigative Function is unavailable, he shall designate this responsibility to an established alternate, who satisfies the formal training and experience for the Superintendent of the Offsite Review and Investigative Function. The responsibilities of the personnel performing this function are stated below. The Offsite Review and Investigative Function shall review:

- 1) The safety evaluations for: (1) changes to procedures, equipment, or systems as described in the safety analysis report, and (2) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question. Proposed changes to the Quality Assurance Program description shall be reviewed and approved by the Manager of Quality Assurance;
- 2) Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;
- 3) Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;
- 4) Proposed changes in Technical Specifications or this Operating License;

ADMINISTRATIVE CONTROLS

OFFSITE (Continued)

- 5) Noncompliance with Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures, or instructions having nuclear safety significance;
- 6) Significant operating abnormalities or deviation from normal and expected performance of plant equipment that affect nuclear safety as referred to it by the Onsite Review and Investigative Function;
- 7) All REPORTABLE EVENTS;
- 8) All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components;
- 9) Review and report findings and recommendations regarding all changes to the Generating Stations Emergency Plan prior to implementation of such change; and
- 10) Review and report findings and recommendations regarding all items referred by the Technical Staff Supervisor, Station Manager, Assistant Vice President and General Manager - Nuclear Stations, and Manager of Quality Assurance.

b. Audit Function

The audit function shall be the responsibility of the Manager of Quality Assurance independent of the Production Department. Such responsibility is delegated to the Director of Quality Assurance (Operations) and the Director of Quality Assurance (Maintenance).

Either of the above, or designated corporate staff or supervision approved by the Manager of Quality Assurance shall approve the audit agenda and checklists, the findings and the report of each audit. Audits shall be performed in accordance with the Company Quality Assurance Program and Procedures. Audits shall be performed to assure that safety-related functions are covered within the period designated below:

- 1) The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;
- 2) The adherence to procedure, training, and qualification of the station staff at least once per 12 months;
- 3) The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or methods of operation that affect nuclear safety at least once per 6 months;
- 4) The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

ADMINISTRATIVE CONTROLS

ONSITE (Continued)

- 3) Review of all proposed changes to the Technical Specifications;
- 4) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety;
- 5) Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Assistant Vice President and General Manager - Nuclear Stations and to the Superintendent of the Offsite Review and Investigative Function;
- 6) Review of all REPORTABLE EVENTS;
- 7) Performance of special reviews and investigations and reports thereon as requested by the Superintendent of the Offsite Review and Investigative Function;
- 8) Review of the Station Security Plan and implementing procedures and submittal of recommended changes to the Station Security Plan to the Director of Corporate Security;
- 9) Review of the Emergency Plan and station implementing procedures and submittal of recommended changes to the Assistant Vice President and General Manager - Nuclear Stations;
- 10) Review of Unit operations to detect potential hazards to nuclear safety;
- 11) Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Assistant Vice President and General Manager - Nuclear Stations and the Superintendent of the Offsite Review and Investigative Function; and
- 12) Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL, and the Radwaste Treatment Systems.
- 13) Review of the Fire Protection Program and implementing instructions and submittal of recommended changes to the Offsite Review and Investigative Function.

c. Authority

The Technical Staff Supervisor is responsible to the Station Manager and shall make recommendations in a timely manner in all areas of review, investigation, and quality control phases of plant maintenance, operation, and administrative procedures relating to facility operations and shall have the authority to request the action necessary to ensure compliance with rules, regulations, and procedures when in his opinion such action is necessary. The Station Manager shall follow such recommendations or select a course

ADMINISTRATIVE CONTROLS

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 Regional Administrator of the NRC Regional Office 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 Final Safety Analysis Report 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 3 months until all three events have been completed.

ANNUAL REPORTS

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.

6.9.1.5 Reports required on an annual basis shall include:

- a. Tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions,* e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be

*This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

ADMINISTRATIVE CONTROLS

REPORTING REQUIREMENTS (Continued)

accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

- b. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. 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.



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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 10
License No. NPF-72

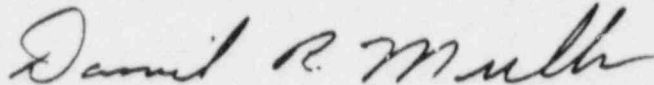
1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Commonwealth Edison Company (the licensee) dated January 18, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specification as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No.10 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION



Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 27, 1988



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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 10
License No. NPF-77

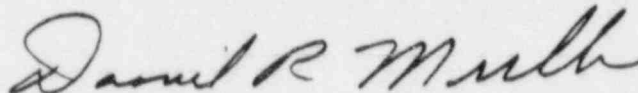
1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Commonwealth Edison Company (the licensee) dated January 18, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specification as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 10 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION



Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 27, 1988

ATTACHMENT TO LICENSE AMENDMENT NOS. 10 AND 10
AND FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77
DOCKET NOS. STN-50-456 AND STN 50-457

Revise Appendix A as follows:

<u>Remove Pages</u>	<u>Insert Pages</u>
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
3/4 4-40	3/4 4-40
3/4 5-1	3/4 5-1
B3/4 4-5	B3/4 4-5
B3/4 4-6	B3/4 4-6
5-6	5-6
6-8	6-8
6-13	6-13
6-18	6-18
	6-18a

REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

- a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and
- b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

- a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and
- b. With the specific activity of the reactor coolant greater than $100/\bar{E}$ microCuries per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

* With T_{avg} greater than or equal to 500°F.

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

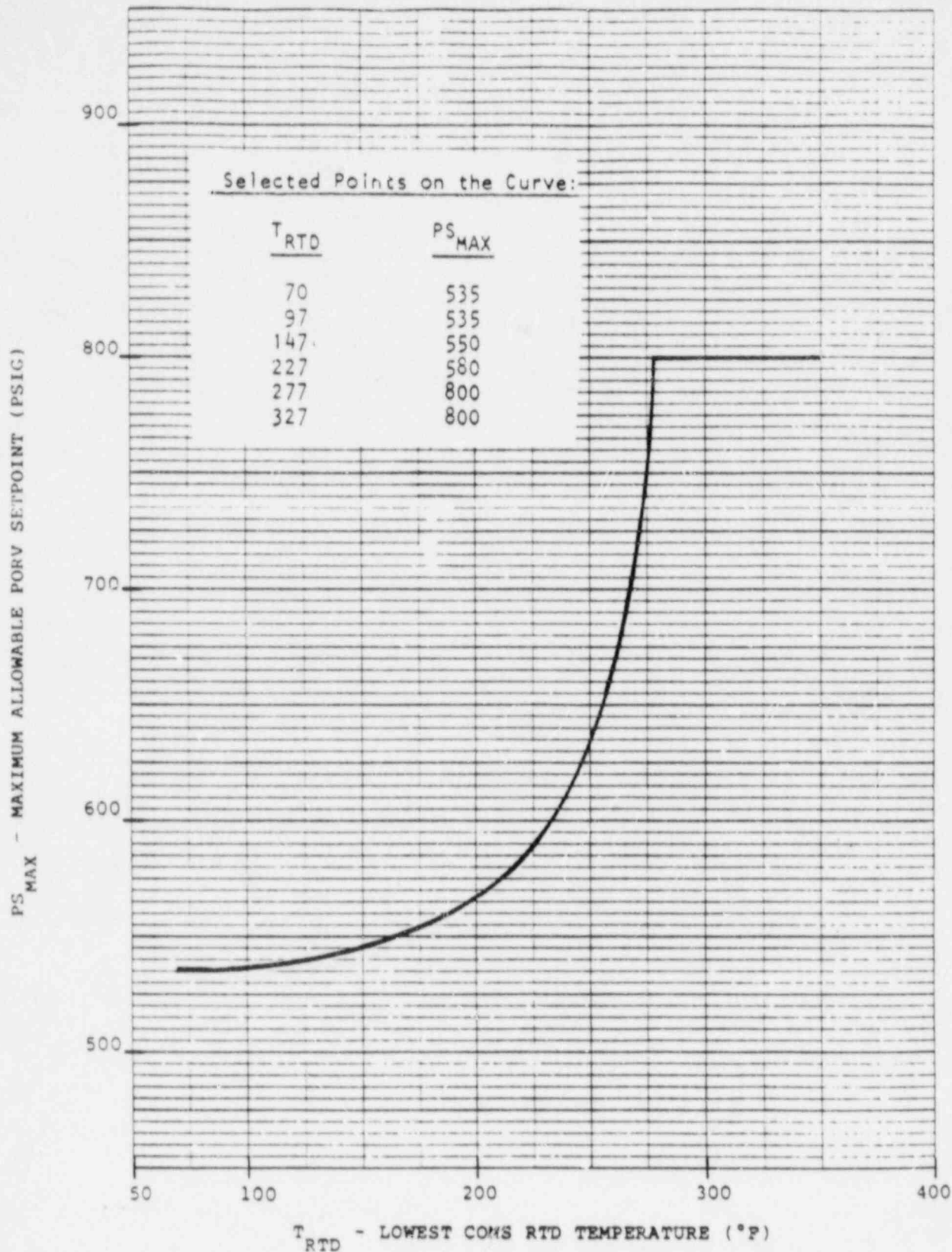


FIGURE 3.4-4

NOMINAL PORV PRESSURE RELIEF SETPOINT VERSUS
RCS TEMPERATURE FOR THE COLD OVERPRESSURE PROTECTION SYSTEM
APPLICABLE UP TO 10 EFY

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

- a. The isolation valve open and power removed,
- b. A contained borated water level of between 31% and 63%,
- c. A boron concentration of between 1900 and 2100 ppm, and
- d. A nitrogen cover-pressure of between 602 and 647 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

- a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

- a. At least once per 12 hours by:
 - 1) Verifying the contained borated water level and nitrogen cover-pressure in the tanks, and
 - 2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

REACTOR COOLANT SYSTEM

BASES

OPERATIONAL LEAKAGE (Continued)

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Braidwood Station, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

REACTOR COOLANT SYSTEM

BASES

SPECIFIC ACTIVITY (Continued)

The sample analysis for determining the gross specific activity and \bar{E} can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is related to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinct window for determination of the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample with typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

<u>COMPONENT</u>	<u>CYCLIC OR TRANSIENT LIMIT</u>	<u>DESIGN CYCLE OR TRANSIENT</u>
Reactor Coolant System	200 heatup cycles at $\leq 100^{\circ}\text{F/h}$ and 200 cooldown cycles at $\leq 100^{\circ}\text{F/h}$.	Heatup cycle - T_{avg} from $\leq 200^{\circ}\text{F}$ to $> 550^{\circ}\text{F}$. Cooldown cycle - T_{avg} from $\geq 550^{\circ}\text{F}$ to $\leq 200^{\circ}\text{F}$.
	200 pressurizer cooldown cycles at $\leq 200^{\circ}\text{F/h}$.	Pressurizer cooldown cycle temperatures from $\geq 650^{\circ}\text{F}$ to $\leq 100^{\circ}\text{F}$.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	$> 15\%$ of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential $> 320^{\circ}\text{F}$.
	200 leak tests.	Pressurized to ≥ 2485 psig.
	10 hydrostatic pressure tests.	Pressurized to ≥ 3107 psig.
	Secondary Coolant System	1 large steam line break.
10 hydrostatic pressure tests.		Pressurized to ≥ 1481 psig.

ADMINISTRATIVE CONTROLS

OFFSITE (Continued)

- 5) Noncompliance with Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures, or instructions having nuclear safety significance;
- 6) Significant operating abnormalities or deviation from normal and expected performance of plant equipment that affect nuclear safety as referred to it by the Onsite Review and Investigative Function;
- 7) ALL REPORTABLE EVENTS;
- 8) All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components;
- 9) Review and report findings and recommendations regarding all changes to the Generating Stations Emergency Plan prior to implementation of such change; and
- 10) Review and report findings and recommendations regarding all items referred by the Technical Staff Supervisor, Station Manager, Assistant Vice President and General Manager of Nuclear Stations, and Manager of Quality Assurance.

b. Audit Function

The audit function shall be the responsibility of the Manager of Quality Assurance independent of the Production Department. Such responsibility is delegated to the Director of Quality Assurance (Operations) and the Director of Quality Assurance (Maintenance).

Either of the above, or designated Corporate Staff or Supervision approved by the Manager of Quality Assurance shall approve the audit agenda and checklists, the findings and the report of each audit. Audits shall be performed in accordance with the Company Quality Assurance Program and Procedures. Audits shall be performed to assure that safety-related functions are covered within the period designated below:

- 1) The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;
- 2) The adherence to procedure, training, and qualification of the station staff at least once per 12 months;
- 3) The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or methods of operation that affect nuclear safety at least once per 6 months;
- 4) The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

ADMINISTRATIVE CONTROLS

ONSITE (Continued)

- 3) Review of all proposed changes to the Technical Specifications;
- 4) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety;
- 5) Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Assistant Vice President and General Manager of Nuclear Stations and to the Superintendent of the Offsite Review and Investigative Function;
- 6) Review of all REPORTABLE EVENTS;
- 7) Performance of special reviews and investigations and reports thereon as requested by the Superintendent of the Offsite Review and Investigative Function;
- 8) Review of the Station Security Plan and implementing procedures and submittal of recommended Security Plan changes to the Director of Corporate Security;
- 9) Review of the Emergency Plan and station implementing procedures and submittal of recommended changes to the Station Security Plan to the Director of Corporate Security;
- 10) Review of Unit operations to detect potential hazards to nuclear safety;
- 11) Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Assistant Vice President and General Manager of Nuclear Stations and the Superintendent of the Offsite Review and Investigative Function; and
- 12) Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL, and the Radwaste Treatment Systems.
- 13) Review of the Fire Protection Program and implementing instructions and submittal of recommended changes to the Offsite Review and Investigative Function.

c. Authority

The Technical Staff Supervisor is responsible to the Station Manager and shall make recommendations in a timely manner in all areas of review, investigation, and quality control phases of plant maintenance, operation, and administrative procedures relating to facility operations and shall have the authority to request the action necessary to ensure compliance with rules, regulations, and procedures when in his opinion such action is necessary. The Station Manager shall follow such recommendations or select a course of action that

ADMINISTRATIVE CONTROLS

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 Regional Administrator of the NRC Regional Office 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 Final Safety Analysis Report 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 3 months until all three events have been completed.

ANNUAL REPORTS

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.

6.9.1.5 Reports required on an annual basis shall include:

- 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,* e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated 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 should be assigned to specific major work functions.

*This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

ADMINISTRATIVE CONTROLS

REPORTING REQUIREMENTS (Continued)

- b. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. 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.