April 13, 1992

Docket Nos. STN 50-454, STN 50-455
STN 50-456, STN 50-457

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Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M80542, M80543, M80544, AND M80545)

The Commission has issued the enclosed Amendment No. 46 to Facility Operating License No. NPF-37 and Amendment No. 46 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 35 to Facility Operating License No. NPF-72 and Amendment No. 35 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated June 10, 1991, and a supplement dated October 17, 1991, consisting of changes to the Technical Specifications (TS).

The amendments change the TS by removing the TS on radioactive effluents and radiological environmental monitoring and adding controls to include them in the Offsite Dose Calculation Manual (ODCM). Specifications on solid radioactive wastes will be relocated to the Process Control Program (PCP). These revisions are in response to Generic Letter 89-01, dated January 31, 1989. Various administrative changes were also made to indicate specification sections that have been deleted and are not being used, and to make Byron and Braidwood TS consistent.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
Original signed by:
Robert M. Pulsifer, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 46 to NPF-37
2. Amendment No. 46 to NPF-66
3. Amendment No. 35 to NPF-72
4. Amendment No. 35 to NPF-77
5. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PDIII-2	PM:PDIII-2	PM:PDIII-2	PM:PDV	PD:PDIII-2	OGC
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DATE	2/ /92	2/19/92	2/20/92	4 /Feb/92	/ /92	2/25/92

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DF01

Mr. Thomas J. Kovach
Commonwealth Edison Company

Byron/Braidwood Power Stations

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 46
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 June 10, 1991 as supplemented on October 17, 1991, 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-37 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 46 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

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 13, 1992

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 46
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 June 10, 1991 as supplemented on October 17, 1991, 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 1;

   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. 46 and revised by Attachment 2 to NPF-66, and the Environmental Protection Plan contained in Appendix B, both of which were 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

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 13, 1992

ATTACHMENT TO LICENSE AMENDMENT NOS. 46 AND 46

FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66

DOCKET NOS. STN 50-454 AND STN 50-455

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf pages identified by an asterisk are provided for convenience.

Remove Pages	Insert Pages
I	I
II	II
V*	V*
VI	VI
VII	VII
VIII	VIII
XIII	XIII
XIV	XIV
XV	XV
XVI	XVI
XVII	XVII
XVIII	XVIII
XIX	XIX
XX	XX
XXI	-
1-3*	1-3*
1-4	1-4
1-5	1-5
1-6	1-6
3/4 3-56	3/4 3-56
3/4 3-57	3/4 3-57
3/4 3-58	3/4 3-58
3/4 3-59	3/4 3-59
3/4 3-60	3/4 3-60
3/4 3-61	3/4 3-61
3/4 3-62	3/4 3-62
3/4 3-63	3/4 3-63
3/4 3-64	3/4 3-64
3/4 3-65	3/4 3-65
3/4 3-66 thru 73	-
3/4 11-1	3/4 11-1
3/4 11-2	3/4 11-2
3/4 11-3	3/4 11-3
3/4 11-4 thru 19	-
3/4 12-1 thru 14	-
B 3/4 3-6	B 3/4 3-6
B 3/4 3-7	B 3/4 3-7
B 3/4 3-8	B 3/4 3-8
B 3/4 3-9	-
B 3/4 7-7	B 3/4 7-7
B 3/4 11-1	B 3/4 11-1
B 3/4 11-2	B 3/4 11-2

Remove Pages	Insert Pages
B 3/4 11-3 thru B 3/4 11-7	-
B 3/4 12-1 thru B 3/4 12-2	-
6-17*	6-17*
-	6-17a
-	6-17b
6-18	6-18
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DEFINITIONS

SECTION | PAGE

1.0 DEFINITIONS

SECTION PAGE

SECTION | PAGE

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.16 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.17 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors or vendors and persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Sections 6.8.4e and f, and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semiannual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.7.

OPERABLE - OPERABILITY

1.19 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATING LIMITS REPORT

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

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM

1.23 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.24 PURGE or PURGING shall be any 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 required to purify the confinement.

QUADRANT POWER TILT RATIO

1.25 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.26 RATED THERMAL POWER shall be a total core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.30 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SLAVE RELAY TEST

1.31 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOLIDIFICATION

1.32 Deleted

SOURCE CHECK

1.33 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.34 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

1.35 THERMAL POWER shall be the total core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.36 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.37 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.38 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

INSTRUMENTATION

MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 (This specification number is not used.)

TABLE 3.3-11

(This table number is not used.)

TABLE 3.3-12

(This table number is not used.)

INSTRUMENTATION

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.8 The Loose-Part Detection System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more Loose-Part Detection System channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each channel of the Loose-Part Detection Systems shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 24 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST except for verification of setpoint at least once per 31 days, and

c. A CHANNEL CALIBRATION at least once per 18 months.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 (This specification number is not used.)

EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The explosive gas monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.2.5 are not exceeded.

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With an explosive gas monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, declare the channel inoperable and take the ACTION shown in Table 3.3-13.

b. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, prepare and submit a Special Report to the Commission within the next 30 days pursuant to Specification 6.9.2 to explain why this inoperability was not corrected in a timely manner.

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

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each explosive gas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and DIGITAL CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-9.

TABLE 3.3-13

EXPLOSIVE GAS MONITORING INSTRUMENTATION#

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. (Not Used)			
2. (Not Used)			
3. Gaseous Waste Management System			
a. Hydrogen Analyzer (OAT-GW8000)	1	**	38
b. Oxygen Analyzer (OAT-GW8003)	1	**	38
c. Waste Gas Compressor Discharge Oxygen Analyzer (OAIT-GW004)	1	***	38

TABLE NOTATIONS

*(not used)
**During WASTE GAS HOLDUP SYSTEM operation.
***During Waste Gas Compressor Operation.
#All instruments required for Unit 1 or Unit 2 operation.

ACTION STATEMENTS

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 24 hours during other operations.

TABLE 4.3-9

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS#

FUNCTIONAL UNIT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. (Not Used)					
2. (Not Used)					
3. Gaseous Waste Management System					
a. Hydrogen Analyzer (OAT-GW8000)	D	N.A.	Q(4)	M	**
b. Oxygen Analyzer (OAT-GW8003)	D	N.A.	Q(5)	M	**
c. Waste Gas Compressor Discharge Oxygen Analyzer (OAIT-GW004)	D	N.A.	Q(5)	M	***

TABLE NOTATIONS

*(Not used)
**During WASTE GAS HOLDUP SYSTEM operation.
***During Waste Gas Compressor Operation.
#All instruments required for Unit 1 or Unit 2 operation.

(1) (Not used)

(2) (Not used)

(3) (Not used)

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing hydrogen and nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing oxygen and nitrogen.

INSTRUMENTATION

HIGH ENERGY LINE BREAK ISOLATION SENSORS

LIMITING CONDITION FOR OPERATION

3.3.3.11 The high energy line break instrumentation shown in Table 3.3-14 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-14.

ACTION:

a. With the number of OPERABLE auxiliary steam isolation instruments less than the Minimum Channels OPERABLE as required by Table 3.3-14, restore the inoperable instrument(s) to OPERABLE status within 7 days, or suspend the supply of auxiliary steam to the Auxiliary Building, or establish a continuous watch in the affected area(s) until the inoperable sensors are restored to OPERABLE status.

b. With the number of OPERABLE steam generator blowdown line isolation instruments less than the Minimum Channels OPERABLE as required by Table 3.3-14, restore the inoperable instrument(s) to OPERABLE status within 7 days, or limit the total steam generator blowdown flow rate to less than or equal to 60 gpm or establish a continuous watch in the affected area(s) until the inoperable sensors are restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.12 Each of the above high energy line break isolation instruments shall be demonstrated OPERABLE by the performance of an ANALOG CHANNEL OPERATIONAL TEST AND CHANNEL CALIBRATION at least once per 18 months.

Table 3.3-14

HIGH ENERGY LINE BREAK INSTRUMENTATION

ISOLATION FUNCTION	INSTRUMENT CHANNEL	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES
1. Auxiliary Steam Isolation	OTS-AS031A OTS-AS032A	1	*
	OTS-AS031B OTS-AS032B	1	*
	OTS-AS031C OTS-AS032C	1	*
	OTS-AS031D OTS-AS032D	1	*
	OTS-AS031E OTS-AS032E	1	*
	OTS-AS031F OTS-AS032F	1	*
2. Steam Generator Blowdown Line Isolation	TS-SD045A TS-SD045B	1	1, 2, 3, 4
	TS-SD046A TS-SD046B	1	1, 2, 3, 4
	TS-SD045C TS-SD045D	1	1, 2, 3, 4
	TS-SD046C TS-SD046D	1	1, 2, 3, 4

*Required when auxiliary steam is being supplied, from any source, to the Auxiliary Building.

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one throttle valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam line(s) or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:

a. During turbine operation at least once per 31 days by direct observation of the movement of the valves below through one complete cycle from the running position:

   1) Four high pressure turbine throttle valves,
   2) Four high pressure turbine governor valves,
   3) Six turbine reheat stop valves, and
   4) Six turbine reheat intercept valves.

b. Within 7 days prior to entering MODE 3 from MODE 4, by cycling each of the 12 extraction steam nonreturn check valves from the closed position,

c. During turbine operation at least once per 31 days by direct observation, of freedom of movement of each of the 12 extraction steam nonreturn check valve weight arms,

d. At least once per 18 months by performance of CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems, and

e. At least once per 40 months by disassembling at least one of each of the valves given in Specifications 4.3.4.2a. and b. above, and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.1 Deleted
3.11.1.2 Deleted
3.11.1.3 Deleted

3.11.1.4 The quantity of radioactive material, excluding tritium and dissolved or entrained noble gases, contained in any outside tanks shall be limited to the following:

a. Primary Water Storage Tank $\leq$ 2000 Curies, and

b. Outside Temporary Tank $\leq$ 10 Curies.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

3/4.11.2 GASEOUS EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the WASTE GAS HOLDUP SYSTEM shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a. above.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM shall be determined to be within the above limits by continuously monitoring the waste gases in the WASTE GAS HOLDUP SYSTEM with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10.

GAS DECAY TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas decay tank shall be limited to less than or equal to 5×10^4 Curies of noble gases (considered as Xe-133 equivalent).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas decay tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and, within 48 hours, reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas decay tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

3/4.3.3.8 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the Reactor Coolant System and avoid or mitigate damage to Reactor Coolant System components. The allowable out-of-service times and Surveillance Requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.9 DELETED

3/4.3.3.10 EXPLOSIVE GAS MONITORING INSTRUMENTATION

The instrumentation includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM.

3/4 3.3.11 HIGH ENERGY LINE BREAK ISOLATION SENSORS

The OPERABILITY of the high energy line break isolation sensors ensures that the capability is available to promptly detect and initiate protective action in the event of a line break. This capability is required to prevent the potential for damage to safety-related systems and structures in the auxiliary building.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment, or structures.

Specification 4.3.4.2a (High Pressure Turbine and Reheat Valves)

These valves isolate large quantities of steam with high potential for delivering energy to the rotor system. The turbine design recognizes this potential in providing rapid action, dual shut off capability in each path, remote testing capability, and a flow path that reduces the effects of changes in flow distribution, load reductions, and thermal transients during testing. The testing intervals are in accordance with the latest manufacturer's recommendations: "Operation and Maintenance Memo 041," Steam Turbine Division, Westinghouse.

TURBINE OVERSPEED PROTECTION (continued)

Specification 4.3.4.2b and c (Extraction Steam Non-Return Check Valves)

These valves are provided to protect the turbine from reflux of steam remaining in the feedwater heater shells and piping following the pressure reduction caused by the actuation of valves in Specification 4.3.4.2a. The quantities of stored steam controlled by these valves are smaller and are divided up into separate heater shells. The feedwater heating system design, including these valves, did not intend routine full stroke testing.

The extraction steam check valves are self closing swing disk non-return valves which shut under the combined effect of gravity and reverse flow of steam. The weight of the disk is partly balanced by a counterweight and lever on the pivot shaft. A spring cylinder acting on the lever assists the start of the automatic closing, but is not intended to close the valve fully against normal steam flow and pressure. In normal operation the spring assist is held clear by air pressure acting on a piston under the spring. The turbine trip system releases the air pressure to assist the closing.

Manual stroking of the extraction steam non-return valves is possible under shutdown conditions by latching the turbine and applying the air pressure to the spring cylinder. It is possible to hear and feel the disk contact the seat solidly. This manual stroking was not provided for in the design but will be done within 7 days prior to entering Mode 3 from Mode 4.

The engineering specifications provided for testing the extraction steam non-return check valves during operation by equalizing the air pressure across the piston in the spring cylinder, permitting the spring to partially close the disk against the steam flow. The rotation of the shaft accompanying the disk closure can be observed by movement of the weight lever. The amount of movement observed in other stations has depended on the extraction steam conditions and valve size, but has been ample to indicate freedom of movement, and this will be verified during startup testing.

Partial stroking demonstrates that the disk system is free at the beginning of the closing stroke where the steam closing forces are smallest. As the disk enters a reverse steam flow the closing forces build up rapidly with progressive closure.

The design of the feedwater heating system is such that full stroke testing of the extraction steam non-return valves during turbine operation involves several penalties without significant additional advantages over partial stroke testing. The motor-operated isolating valve must be closed on an individual heater. Heater stages 1, 2, 3, and 4 are arranged in three parallel strings with cascaded drains in each string and heater stages 5, 6 and 7 are similarly arranged in two parallel strings. An entire string is taken out of service, isolated, and bypassed for maintenance. Isolating the extraction steam to a single intermediate heater involves several complications.

TURBINE OVERSPEED PROTECTION (continued)

The motor operated valves are too large for routine manual operation, do not have bypasses to allow controlled warmup conditions, stroke quickly (about 15 seconds), and are intended for turbine protection against heater flooding. A comparison of the thermal capacity of a heater and the rate of heat transfer to the flowing condensate or feedwater shows that cycling an extraction steam isolating valve would cause rapid cooling and heating transients.

Isolating the steam to a top heater drops the feedwater temperature to the steam generators. Isolating the steam to an intermediate heater causes the next heater to assume the heating load, approximately doubling the steam demand and drain flow, and nearly quadrupling the potential for erosion and vibration in the affected heater and piping. The shell pressure collapses in the isolated heater causing insufficient head to discharge the cascading drains to the next lower heater. Rapid action of the emergency drain control is required to prevent flooding, with the potential for flashing in the drain cooler section from pressure decay.

Isolating a heater degrades the cycle thermal performance, requiring a corresponding drop in electrical output for the same reactor thermal power.

Partial closing of the extraction steam non-return check valves with the installed test provisions demonstrates freedom of movement while avoiding transient states. A 31-day interval will be adequate since it is likely that sticking conditions would develop during shutdown conditions rather than in operation.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.10 (This specification number is not used.)

3/4.7.11 (This specification number is not used.)

3/4.7.12 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 DELETED

3/4.11.1.2 DELETED

3/4.11.1.3 DELETED

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3.4.11.2 GASEOUS EFFLUENTS

3.4.11.2.1 DELETED

3.4.11.2.2 DELETED

3.4.11.2.3 DELETED

3/4.11.2.4 DELETED

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS DECAY TANKS

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

PROCEDURES AND PROGRAMS (Continued)

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1) Training of personnel,

2) Procedures for monitoring, and

3) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables,

2) Identification of the procedures used to measure the values of the critical variables,

3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

4) Procedures for the recording and management of data,

5) Procedures defining corrective action for all off-control point chemistry conditions, and

6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Post-accident Sampling

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

1) Training of personnel,

2) Procedures for sampling and analysis, and

3) Provisions for maintenance of sampling and analysis equipment.

PROCEDURES AND PROGRAMS (Continued)

e. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by station procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1,

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

PROCEDURES AND PROGRAMS (Continued)

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, and

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

f. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

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 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.

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.

REPORTING REQUIREMENTS (Continued)

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.6 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.7 The Semiannual Radioactive Effluent Release Report 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. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

MONTHLY OPERATING REPORT

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or RCS safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

OPERATING LIMITS REPORT

6.9.1.9 Operating limits shall be established and documented in the OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle. The analytical methods used to determine the operating limits shall be those previously reviewed and approved by the NRC in Topical Reports: 1) WCAP 9272-P-A "Westinghouse Reload Safety Evaluations Methodology" dated July 1985, 2) WCAP-8385 "Power Distribution Control and Load Following Procedures" dated September 1974, 3) NFSR-0016 "Benchmark of PWR Nuclear Design Methods" dated July 1983, and/or 4) NFSR-0081 "Benchmark of PWR Nuclear Design Methods Using the PHOENIX-P and ANC Computer Codes" dated July 1990. The operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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*A single submittal may be made for a multi-unit station.

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

CRITICALITY ANALYSIS OF BYRON AND BRAIDWOOD STATION FUEL STORAGE RACKS

6.9.1.10 Fuel enrichment limits for storage shall be established and documented in the CRITICALITY ANALYSIS OF BYRON AND BRAIDWOOD STATION FUEL STORAGE RACKS. The analytical methods used to determine the maximum fuel enrichments shall be those previously reviewed and approved by the NRC in "CRITICALITY ANALYSIS OF BYRON AND BRAIDWOOD STATION FUEL STORAGE RACKS". The fuel enrichment limits for storage shall be determined so that all applicable limits (e.g., subcriticality) of the safety analysis are met.

The CRITICALITY ANALYSIS OF BYRON AND BRAIDWOOD STATION FUEL STORAGE RACKS report shall be provided upon issuance of any changes, to the NRC Document Control Desk, with copies to the Regional Administrator and the Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

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 5 years:

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

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

c. All REPORTABLE EVENTS;

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

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

f. Records of radioactive shipments;

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

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

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

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

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

RECORD RETENTION (Continued)

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

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

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

f. Records of reactor tests and experiments;

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

h. Records of in-service inspections performed pursuant to these Technical Specifications;

i. Records of Quality Assurance activities required by the QA Program;

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of meetings and results of reviews and audits performed by the Offsite Review and Investigative Function and the Onsite Review and Investigative Function;

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.8 including the date at which the service life commences and associated installation and maintenance records;

m. Records of secondary water sampling and water quality;

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

o. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to Paragraph 20.203(c)(5) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less than 1000 mR/hr at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area; or

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the Radiation Work Permit.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area. During emergency situations which involve personnel injury or actions taken to prevent major equipment damage, continuous surveillance and radiation monitoring of the work area by a qualified individual may be substituted for the routine RWP procedure.

HIGH RADIATION AREA (Continued)

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded (by a more substantial obstacle than rope), conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2o. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

   2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the Onsite Review and Investigative Function (Onsite Review) and the approval of the Station Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2o. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

   2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the Onsite Review and Investigative Function and the approval of the Station Manager on the date specified by the Onsite Review and Investigative Function.

OFFSITE DOSE CALCULATION MANUAL (ODCM) (Continued)

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Semiannual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made effective. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 35
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 June 10, 1991 as supplemented on October 17, 1991, 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-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 35 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

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 13, 1992

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 35
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 June 10, 1991 as supplemented on October 17, 1991, 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 1;

   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-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 35 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 if its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 13, 1992

ATTACHMENT TO LICENSE AMENDMENT NOS. 35 AND 35

FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

DOCKET NOS. STN 50-456 AND STN 50-457

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

Remove Pages	Insert Pages
I	I
II	II
V*	V*
VI	VI
VII	VII
XI*	XI*
XII	XII
XIII	XIII
XIV	XIV
XV	XV
XVI	XVI
XVII	XVII
XVIII	XVIII
XIX	XIX
XX	XX
XXI	-
1-3*	1-3*
1-4	1-4
1-5	1-5
1-6	1-6
3/4 3-57	3/4 3-57
3/4 3-58*	3/4 3-58*
3/4 3-59	3/4 3-59
3/4 3-60	3/4 3-60
3/4 3-61	3/4 3-61
3/4 3-62	3/4 3-62
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3/4 3-64	3/4 3-64
3/4 3-65	3/4 3-65
3/4 3-66 thru 75	-
3/4 11-1	3/4 11-1
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3/4 11-3	3/4 11-3
3/4 11-4 thru 19	-
3/4 12-1 thru 12-14	-
B 3/4 3-5*	B 3/4 3-5*
B 3/4 3-6	B 3/4 3-6
B 3/4 3-7	B 3/4 3-7
B 3/4 3-8	B 3/4 3-8
B 3/4 11-1	B 3/4 11-1
B 3/4 11-2	B 3/4 11-2
B 3/4 11-3 thru B 3/4 11-7	-
B 3/4 12-1 thru B 3/4 12-2	-

Remove Pages	Insert Pages
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DEFINITIONS

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1.0 DEFINITIONS	

SECTION PAGE

SECTION PAGE

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.12 E shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.16 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.17 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors or vendors and persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Sections 6.8.4.e and f, and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semiannual Radioactive Effluent Release Reports required by Specification 6.9.1.6 and 6.9.1.7.

OPERABLE - OPERABILITY

1.19 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATING LIMITS REPORT

1.19.a The OPERATING LIMITS REPORT is the unit-specific document that provides operating limits for the current operating reload cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant Operation within these operating limits is addressed in individual specifications.

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM

1.23 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.24 PURGE or PURGING shall be any 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 required to purify the confinement.

QUADRANT POWER TILT RATIO

1.25 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.26 RATED THERMAL POWER shall be a total core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.30 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SLAVE RELAY TEST

1.31 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOLIDIFICATION

1.32 Deleted

SOURCE CHECK

1.33 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.34 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

1.35 THERMAL POWER shall be the total core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.36 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.37 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.38 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

TABLE 3.3-11

(This table number is not used.)

Table 3.3-12

(This table number is not used.)

INSTRUMENTATION

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.8 The Loose-Part Detection System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more Loose-Part Detection System channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each channel of the Loose-Part Detection Systems shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 24 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST except for verification of setpoint at least once per 31 days, and

c. A CHANNEL CALIBRATION at least once per 18 months.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 (This specification number is not used.)

EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The explosive gas monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.2.5 are not exceeded.

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With an explosive gas monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, declare the channel inoperable and take the ACTION shown in Table 3.3-13.

b. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, prepare and submit a Special Report to the Commission within the next 30 days pursuant to Specification 6.9.2 to explain why this inoperability was not corrected in a timely manner.

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

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each explosive gas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and DIGITAL CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-9.

TABLE 3.3-13

EXPLOSIVE GAS MONITORING INSTRUMENTATION#

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. (Not Used)			
2. (Not Used)			
3. Gaseous Waste Management System			
a. Hydrogen Analyzer (OAT-GW8000)	1	**	38
b. Oxygen Analyzer (OAT-GW8003)	1	**	38
c. Waste Gas Compressor Discharge Oxygen Analyzer (OAIT-GW004)	1	***	38

TABLE NOTATIONS

*(Not used)
**During WASTE GAS HOLDUP SYSTEM operation.
***During Waste Gas Compressor Operation.
#All instruments required for Unit 1 or Unit 2 operation.

ACTION STATEMENTS

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 24 hours during other operations.

TABLE 4.3-9

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS#

FUNCTIONAL UNIT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. (Not Used)					
2. (Not Used)					
3. Gaseous Waste Management System					
a. Hydrogen Analyzer (OAT-GW8000)	D	N.A.	Q(4)	M	**
b. Oxygen Analyzer (OAT-GW8003)	D	N.A.	Q(5)	M	**
c. Waste Gas Compressor Discharge Oxygen Analyzer (OAIT-GW004)	D	N.A.	Q(5)	M	***

TABLE NOTATIONS

*(Not used)
**During WASTE GAS HOLDUP SYSTEM operation.
***During Waste Gas Compressor Operation.
#All instruments required for Unit 1 or Unit 2 operation.

(1) (Not Used)

(2) (Not Used)

(3) (Not Used)

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing hydrogen and nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing oxygen and nitrogen.

INSTRUMENTATION

HIGH ENERGY LINE BREAK ISOLATION SENSORS

LIMITING CONDITION FOR OPERATION

3.3.3.11 The high energy line break instrumentation shown in Table 3.3-14 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-14

ACTION:

a. With the number of OPERABLE auxiliary steam isolation instruments less than the Minimum Channels OPERABLE as required by Table 3.3-14, restore the inoperable instrument(s) to OPERABLE status within 7 days, or suspend the supply of auxiliary steam to the Auxiliary Building, or establish a continuous watch in the affected area(s) until the inoperable sensors are restored to OPERABLE status.

b. With the number of OPERABLE steam generator blowdown line isolation instruments less than the Minimum Channels OPERABLE as required by Table 3.3-14, restore the inoperable instrument(s) to OPERABLE status within 7 days, or limit the total steam generator blowdown flow rate to less than or equal to 60 gpm or establish a continuous watch in the affected area(s) until the inoperable sensors are restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.12 Each of the above high energy line break isolation instruments shall be demonstrated OPERABLE by the performance of an ANALOG CHANNEL OPERATIONAL TEST and CHANNEL CALIBRATION at least once per 18 months.

TABLE 3.3-14

HIGH ENERGY LINE BREAK INSTRUMENTATION

ISOLATION FUNCTION	INSTRUMENT CHANNEL	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES
1. Auxiliary Steam Isolation	OTS-AS031A OTS-AS032A	1	*
	OTS-AS031B OTS-AS032B	1	*
	OTS-AS031C OTS-AS032C	1	*
	OTS-AS031D OTS-AS032D	1	*
	OTS-AS031E OTS-AS032E	1	*
	OTS-AS031F OTS-AS032F	1	*
2. Steam Generator Blowdown Line Isolation	TS-SD045A TS-SD045B	1	1, 2, 3, 4
	TS-SD046A TS-SD046B	1	1, 2, 3, 4
	TS-SD045C TS-SD045D	1	1, 2, 3, 4
	TS-SD046C TS-SD046D	1	1, 2, 3, 4

* Required when auxiliary steam is being supplied, from any source, to the Auxiliary Building.

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one throttle valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam line(s) or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:

a. During turbine operation at least once per 31 days by direct observation of the movement of the valves below through one complete cycle from the running position:

   1) Four high pressure turbine throttle valves,
   2) Four high pressure turbine governor valves,
   3) Six turbine reheat stop valves, and
   4) Six turbine reheat intercept valves.

b. Within 7 days prior to entering MODE 3 from MODE 4, by cycling each of the 12 extraction steam nonreturn check valves from the closed position,

c. During turbine operation at least once per 31 days by direct observation, of freedom of movement of each of the 12 extraction steam nonreturn check valve weight arms,

d. At least once per 18 months by performance of CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems, and

e. At least once per 40 months by disassembling at least one of each of the valves given in Specifications 4.3.4.2a. and b. above, and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.1 Deleted
3.11.1.2 Deleted
3.11.1.3 Deleted

3.11.1.4 The quantity of radioactive material, excluding tritium and dissolved or entrained noble gases, contained in any outside tanks shall be limited to the following:

a. Primary Water Storage Tank $\leq$ 2000 Curies, and

b. Outside Temporary Tank $\leq$ 10 Curies.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the WASTE GAS HOLDUP SYSTEM shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a. above.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM shall be determined to be within the above limits by continuously monitoring the waste gases in the WASTE GAS HOLDUP SYSTEM with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10.

RADIOACTIVE EFFLUENTS

GAS DECAY TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas decay tank shall be limited to less than or equal to 5×10^4 Curies of noble gases (considered as Xe-133 equivalent).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas decay tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and, within 48 hours, reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas decay tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

SEISMIC INSTRUMENTATION (Continued)

The response spectrum analyzer computes the response spectrum of the event for two sensor locations, compares it to the design response spectra of the plant, and indicates whether the event exceeded the operating basis earthquake criteria or the safe shutdown earthquake criteria.

This instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquake," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG 0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.7 (This specification number is not used.)

3/4.3.3.8 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the Reactor Coolant System and avoid or mitigate damage to Reactor Coolant System components. The allowable out-of-service times and Surveillance Requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.9 DELETED

3/4.3.3.10 EXPLOSIVE GAS MONITORING INSTRUMENTATION

The instrumentation includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM.

3/4.3.3.11 HIGH ENERGY LINE BREAK ISOLATION SENSORS

The OPERABILITY of the high energy line break isolation sensors ensures that the capability is available to promptly detect and initiate protective action in the event of a line break. This capability is required to prevent the potential for damage to safety-related systems and structures in the auxiliary building.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment, or structures.

Specification 4.3.4.2a (High Pressure Turbine and Reheat Valves)

These valves isolate large quantities of steam with high potential for delivering energy to the rotor system. The turbine design recognizes this potential in providing rapid action, dual shut off capability in each path, remote testing capability, and a flow path that reduces the effects of changes in flow distribution, load reductions, and thermal transients during testing. The testing intervals are in accordance with the latest manufacturer's recommendations: "Operation and Maintenance Memo 041," Steam Turbine Division, Westinghouse.

TURBINE OVERSPEED PROTECTION (Continued)

Specification 4.3.4.2b and c (Extraction Steam Non-Return Check Valves)

These valves are provided to protect the turbine from reflux of steam remaining in the feedwater heater shells and piping following the pressure reduction caused by the actuation of valves in Specification 4.3.4.2a. The quantities of stored steam controlled by these valves are smaller and are divided up into separate heater shells. The feedwater heating system design, including these valves, did not intend routine full stroke testing.

The extraction steam check valves are self closing swing disk non-return valves which shut under the combined effect of gravity and reverse flow of steam. The weight of the disk is partly balanced by a counterweight and lever on the pivot shaft. A spring cylinder acting on the lever assists the start of the automatic closing, but is not intended to close the valve fully against normal steam flow and pressure. In normal operation the spring assist is held clear by air pressure acting on a piston under the spring. The turbine trip system releases the air pressure to assist the closing.

Manual stroking of the extraction steam non-return valves is possible under shutdown conditions by latching the turbine and applying the air pressure to the spring cylinder. It is possible to hear and feel the disk contact the seat solidly. This manual stroking was not provided for in the design but will be done within 7 days prior to entering Mode 3 from Mode 4.

The engineering specifications provided for testing the extraction steam non-return check valves during operation by equalizing the air pressure across the piston in the spring cylinder, permitting the spring to partially close the disk against the steam flow. The rotation of the shaft accompanying the disk closure can be observed by movement of the weight lever. The amount of movement observed in other stations has depended on the extraction steam conditions and valve size, but has been ample to indicate freedom of movement, and this will be verified during startup testing.

Partial stroking demonstrates that the disk system is free at the beginning of the closing stroke where the steam closing forces are smallest. As the disk enters a reverse steam flow the closing forces build up rapidly with progressive closure.

The design of the feedwater heating system is such that full stroke testing of the extraction steam non-return valves during turbine operation involves several penalties without significant additional advantages over partial stroke testing. The motor-operated isolating valve must be closed on an individual heater. Heater stages 1, 2, 3, and 4 are arranged in three parallel strings with cascaded drains in each string and heater stages 5, 6 and 7 are similarly arranged in two parallel strings. An entire string is taken out of service, isolated, and bypassed for maintenance. Isolating the extraction steam to a single intermediate heater involves several complications.

TURBINE OVERSPEED PROTECTION (Continued)

The motor operated valves are too large for routine manual operation, do not have bypasses to allow controlled warmup conditions, stroke quickly (about 15 seconds), and are intended for turbine protection against heater flooding. A comparison of the thermal capacity of a heater and the rate of heat transfer to the flowing condensate or feedwater shows that cycling an extraction steam isolating valve would cause rapid cooling and heating transients.

Isolating the steam to a top heater drops the feedwater temperature to the steam generators. Isolating the steam to an intermediate heater causes the next heater to assume the heating load, approximately doubling the steam demand and drain flow, and nearly quadrupling the potential for erosion and vibration in the affected heater and piping. The shell pressure collapses in the isolated heater causing insufficient head to discharge the cascading drains to the next lower heater. Rapid action of the emergency drain control is required to prevent flooding, with the potential for flashing in the drain cooler section from pressure decay.

Isolating a heater degrades the cycle thermal performance, requiring a corresponding drop in electrical output for the same reactor thermal power.

Partial closing of the extraction steam non-return check valves with the installed test provisions demonstrates freedom of movement while avoiding transient states. A 31-day interval will be adequate since it is likely that sticking conditions would develop during shutdown conditions rather than in operation.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 DELETED

3/4.11.1.2 DELETED

3/4.11.1.3 DELETED

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DELETED

3/4.11.2.2 DELETED

3/4.11.2.3 DELETED

3/4.11.2.4 DELETED

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS DECAY TANKS

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

PROCEDURES AND PROGRAMS (Continued)

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1) Training of personnel,

2) Procedures for monitoring, and

3) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables,

2) Identification of the procedures used to measure the values of the critical variables,

3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

4) Procedures for the recording and management of data,

5) Procedures defining corrective action for all off-control point chemistry conditions, and

6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Post-accident Sampling

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

1) Training of personnel,

2) Procedures for sampling and analysis, and

3) Provisions for maintenance of sampling and analysis equipment.

PROCEDURES AND PROGRAMS (Continued)

e. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by station procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1,

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

PROCEDURES AND PROGRAMS (Continued)

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, and

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

f. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

REPORTING REQUIREMENTS (Continued)

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.6 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.7 The Semiannual Radioactive Effluent Release Report 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. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

MONTHLY OPERATING REPORT

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or RCS safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

OPERATING LIMITS REPORT

6.9.1.9 Operating limits shall be established and documented in the OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle. The analytical methods used to determine the operating limits shall be those previously reviewed and approved by the NRC in Topical Reports: 1) WCAP 9272-P-A "Westinghouse Reload Safety Evaluations Methodology" dated July 1985, 2) WCAP-8385 "Power Distribution Control and Load Following Procedures" dated September 1974, 3) NFSR-0016 "Benchmark of PWR Nuclear Design Methods" dated July 1983, and/or 4) NFSR-0081 "Benchmark of PWR Nuclear Design Methods Using the PHOENIX-P and ANC Computer Codes" dated July 1990. The operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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

CRITICALITY ANALYSIS OF BYRON AND BRAIDWOOD STATION FUEL STORAGE RACKS

6.9.1.10 Fuel enrichment limits for storage shall be established and documented in the CRITICALITY ANALYSIS OF BYRON AND BRAIDWOOD STATION FUEL STORAGE RACKS. The analytical methods used to determine the maximum fuel enrichments shall be those previously reviewed and approved by the NRC in "CRITICALITY ANALYSIS OF BYRON AND BRAIDWOOD STATION FUEL STORAGE RACKS." The fuel enrichment limits for storage shall be determined so that all applicable limits (e.g., subcriticality) of the safety analysis are met.

The CRITICALITY ANALYSIS OF BYRON AND BRAIDWOOD STATION FUEL STORAGE RACKS report shall be provided upon issuance of any changes, to the NRC Document Control Desk, with copies to the Regional Administrator and the Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

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 5 years:

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

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

c. All REPORTABLE EVENTS;

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

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

f. Records of radioactive shipments;

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

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

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

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

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

RECORD RETENTION (Continued)

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

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

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

f. Records of reactor tests and experiments;

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

h. Records of in-service inspections performed pursuant to these Technical Specifications;

i. Records of Quality Assurance activities required by the QA Program;

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of meetings and results of reviews and audits performed by the Offsite Review and Investigative Function and the Onsite Review and Investigative Function;

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.8 including the date at which the service life commences and associated installation and maintenance records;

m. Records of secondary water sampling and water quality;

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

o. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to Paragraph 20.203(c)(5) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less than 1000 mR/hr at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area; or

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the Radiation Work Permit.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area. During emergency situations which involve personnel injury or actions taken to prevent major equipment damage, continuous surveillance and radiation monitoring of the work area by a qualified individual may be substituted for the routine RWP procedure.

HIGH RADIATION AREA (Continued)

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded (by a more substantial obstacle than rope), conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2o. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

   2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the Onsite Review and Investigative Function (Onsite Review) and the approval of the Station Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2o. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

   2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the Onsite Review and Investigative Function and the approval of the Station Manager on the date specified by the Onsite Review and Investigative Function.

OFFSITE DOSE CALCULATION MANUAL (ODCM) (Continued)

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Semiannual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made effective. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 46 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 46 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 35 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 35 TO FACILITY OPERATING LICENSE NO. NPF-77

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNIT NOS. 1 AND 2

BRAIDWOOD STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456 AND STN 50-457

1.0 INTRODUCTION

By letter dated June 10, 1991, as supplemented on October 17, 1991, Commonwealth Edison Company, (CECo, the licensee) proposed to incorporate programmatic controls for radiological effluents and radiological environmental monitoring in the Administrative Controls section of the Technical Specifications (TS) consistent with the requirements of 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50. At the same time, the licensee proposed to transfer the procedural details of the Radiological Effluent Technical Specifications (RETS) from the TS to the Offsite Dose Calculation Manual (ODCM) or to the Process Control Program (PCP) for solid radioactive wastes as appropriate. With these changes, the specifications related to RETS reporting requirements were simplified. Finally, changes to the definitions of the ODCM and PCP were proposed consistent with these changes. Guidance on these proposed changes was provided to all power reactor licensees and applicants by Generic Letter (GL) 89-01 dated January 31, 1989. The October 17, 1991, submittal provided clarifying information that did not change the initial proposed no significant hazards consideration.

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided in GL 89-01 and are addressed below.

(1) The licensee has proposed to incorporate programmatic controls for radioactive effluents and radiological environmental monitoring in TS 6.8.4 "Procedures and Programs," of the TS as noted in the guidance provided in GL 89-01. The programmatic controls ensure that programs

are established, implemented, and maintained to ensure that operating procedures are provided to control radioactive effluents consistent with the requirements of 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50.

(2) The licensee has confirmed that the detailed procedural requirements addressing Limiting Conditions for Operation, their applicability, remedial actions, associated surveillance requirements, or reporting requirements for the following specifications have been prepared to implement the relocation of these procedural details to the ODCM or PCP. These changes to the ODCM and PCP have been prepared in accordance with the new Administrative Controls in the TS on changes to the ODCM and PCP so that they will be implemented in the ODCM or PCP when this amendment is issued.

SPECIFICATION	TITLE
3/4.3.3.9	RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION
3/4.3.3.10	RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
3/4.11.1.1	RADIOACTIVE EFFLUENTS: LIQUID EFFLUENTS CONCENTRATION
3/4.11.1.2	RADIOACTIVE EFFLUENTS: LIQUID EFFLUENTS DOSE
3/4.11.1.3	RADIOACTIVE EFFLUENTS: LIQUID RADWASTE TREATMENT SYSTEM
3/4.11.2.1	RADIOACTIVE EFFLUENTS: GASEOUS EFFLUENTS DOSE RATE
3/4.11.2.2	RADIOACTIVE EFFLUENTS: DOSE - NOBLE GASES
3/4.11.2.3	RADIOACTIVE EFFLUENTS: DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM
3/4.11.2.4	RADIOACTIVE EFFLUENTS: GASEOUS RADWASTE TREATMENT SYSTEM
3/4.11.3	RADIOACTIVE EFFLUENTS: SOLID RADIOACTIVE WASTES
3/4.11.4	RADIOACTIVE EFFLUENTS: TOTAL DOSE
3/4.12.1	RADIOACTIVE ENVIRONMENTAL MONITORING: MONITORING PROGRAM
3/4.12.2	RADIOACTIVE ENVIRONMENTAL MONITORING: LAND USE CENSUS
3/4.12.3	RADIOACTIVE ENVIRONMENTAL MONITORING: INTERLABORATORY COMPARISON PROGRAM
6.9.1.6	ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.7 SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

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

These procedural details that have been removed from the TS are not required by the Commission's regulations to be included in TS. They have been prepared for incorporation in the ODCM or PCP upon issuance of this license amendment and may be subsequently changed by the licensee without prior NRC approval. Changes to the ODCM and PCP are documented and will be retained for the duration of the operating license in accordance with TS 6.10.2o.

(3) The licensee has proposed replacing the existing specifications in the Administrative Controls section of the TS for the Annual Radiological Environmental Operating Report, TS 6.9.1.6 for the Semiannual Radioactive Effluent Release Report, TS 6.9.1.7, for the Process Control Program, TS 6.13, and for the Offsite Dose Calculation Manual, TS 6.14, with the updated TSs that were provided in GL 89-01.

The following TS that are included under the heading of Radioactive Effluents have been retained in the TS. This is in accordance with the guidance of GL 89-01.

SPECIFICATION	TITLE
3/4.3.3.10	EXPLOSIVE GAS MONITORING INSTRUMENTATION (Retained existing requirements of this specification)
3/4.11.1.4	LIQUID HOLDUP TANKS
3/4.11.2.5	EXPLOSIVE GAS MIXTURE
3/4.11.2.6	GAS STORAGE TANKS

Several administrative changes were made to the pages affected by this request. Besides page numbers changing due to deletions of some of these specifications, the following administration changes were made:

(1) Sections 3/4.9.6, 3/4.9.7, and 3/4.9.8 in the Braidwood TS index was repeated on pages XVII and XVIII. These sections were deleted on page XVII to make it consistent with the Byron TS.

(2) Added definitions 1.9a and 1.19a to index page. Also changed page numbers for definitions 1.23, 1.30 and 1.31 to reflect the text.

(3) "Motor-Operated Valves Thermal Overload Protection Devices" on top of index page XII for Braidwood was deleted because it was a repeat from page XI.

(4) On index page XVIII in Byron and Braidwood TS for section 3/4.9.12, the word "System" was deleted to match the title of section 3/4.9.12 in the TS on page B3/4 9-3.

(5) On index page XIX in Byron and Braidwood TS for Table 5.7-1, an "s" was added to the words Limit to reflect Table 5.7-1 in the TS on page 5-6.

(6) Deleted "Table 6.2-1a (This Figure Not Used)" from Braidwood index page XX and deleted page 6-5a from the TS to match the Byron TS.

(7) On Byron TS index page VIII Figure 3.4-4b was added and an "a" was added to the titles of Figures 3.4-3a and 3.4-4a. This was done to reflect the TS text.

(8) In the Byron and Braidwood TS index for specification 6.9.1., Routine Reports, "Radical Peaking Factor Limit Report" was deleted and "Operating Limits Report" and "Criticality Analysis of Byron and Braidwood Station Fuel Storage Racks" were added to reflect TS Section 6.9.1.

(9) Corrected page number on Byron index for specification 3/4.7.9

(10) Added TS Sections 3/4.7.10 and 3/4 7.11 to Byron TS and the index to indicate these sections are not being used. This was done to be consistent with the Braidwood TS and to eliminate confusion.

(11) Moved three lines from Byron TS page 6-18a to 6-18 to make Byron consistent with Braidwood TS.

(12) Moved TABLE 3.4-1 and specification 3/4.4.7 in Byron index from page VIII to page VII. Also added "High Energy Line Break Isolating Sensors" after TABLE 4.3-9 in page VII. This was done to make Byron and Braidwood TS consistent.

(13) Added specification 3.3.3.7 to the Byron TS to indicate specification is not used. Also added Table 3.3-11 (this table number is not being used) in the Byron and Braidwood index. This was done to provide consistency between Byron and Braidwood TS and to eliminate possible confusion of a missing specification.

(14) Braidwood TS 6.9.1.9, on page 6-19, the word PHOENIX was capitalized to make it consistent with Byron TS.

On the basis of the above, the staff finds that the changes included in the proposed TS amendment request are consistent with the guidance provided in GL 89-01. Because the control of radioactive effluents continues to be limited in accordance with operating procedures that must satisfy the regulatory requirements of 10 CFR Part 50, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds the proposed changes acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes in recordkeeping, reporting, or administrative procedures and program requirements. The Commission made proposed determinations that the amendments involve no significant hazards consideration, which were published in the Federal Register (56 FR 37578) on August 7, 1991. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

On the basis of the considerations discussed above, the staff concludes that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will no be inimical to the common defense and security or to the health and safety of the public.

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