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

FEB 13 1986

Docket Nos.: STN 50-454
and STN 50-455

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

SUBJECT: AMENDMENT TO BYRON TECHNICAL SPECIFICATIONS - CORRECT TYPOGRAPHICAL AND GRAMMATICAL ERRORS

The Commission has issued the enclosed Amendment No. 2 to the Facility Operating License NPF-37 for Byron Station, Unit 1. This amendment also applies to the Technical Specifications for Byron Station, Unit 2, although Unit 2 does not have an operating license. Byron Station, Units 1 and 2, have common Technical Specifications (NUREG-1113). This amendment is in response to your application, dated September 27, 1985.

The amendment approves changes to the Technical Specifications to correct typographical and grammatical errors on six pages.

The staff has determined that the amendment involves no increase in the amounts, and no change in the type, of any effluents that may be released offsite and that there is no increase in individual or cumulative occupational radiation exposure. A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration and Opportunity for Hearing related to the requested action was published in the Federal Register on October 23, 1985 (50 FR 43022). The State of Illinois Department of Nuclear Safety, by letter dated October 22, 1985, stated that no adverse findings exist as a result of its review of the proposed amendment. No requests for hearing were received. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Pursuant to 10CFR 51.22 (b) no environmental impact statement nor environment assessment need be prepared in connection with the issuance of this amendment.

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

This action will appear in the Commission's bi-weekly notice publication in the Federal Register.

Sincerely,

/S/

Vincent S. Noonan, Director
PWR Project Directorate #5
Division of PWR Licensing-A

cc: See next page

OFC	:PD#5	:PD#5	:PD#5	: RP	: OELD No legal objection.	:
NAME	:LOlshan	:MRushbrook	:VSNoonan	: VBenaroya	: S H Lewis	:
DATE	: 1 /7 /86	: 1 /7 /86	: 1 /17 /86	: 1/17/86	: ~~1/27/86~~ 2/7/86	:

OFFICIAL RECORD COPY

Mr. Dennis L. Farrar
Commonwealth Edison Company

Byron Station
Units 1 and 2

cc:
Mr. William Kortier
Atomic Power Distribution
Westinghouse Electric Corporation
Post Office Box 355
Pittsburgh, Pennsylvania 15230

Michael Miller
Isham, Lincoln & Beale
One First National Plaza
42nd Floor
Chicago, Illinois 60603

Mrs. Phillip B. Johnson
1907 Stratford Lane
Rockford, Illinois 61107

Dr. Bruce von Zellen
Department of Biological Sciences
Northern Illinois University
DeKalb, Illinois 61107

Mr. Edward R. Crass
Nuclear Safeguards & Licensing
Sargent & Lundy Engineers
55 East Monroe Street
Chicago, Illinois 60603

Mr. Julian Hinds
U. S. Nuclear Regulatory Commission
Byron/Resident Inspectors Offices
4448 German Church Road
Byron, Illinois 61010

Mr. Michael C. Parker, Chief
Division of Engineering
Illinois Department of
Nuclear Safety
1035 Outer Park Drive
Springfield, Illinois 62704

Ms. Diane Chavez
528 Gregory Street
Rockford, Illinois 61108

Regional Administrator, Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Joseph Gallo, Esq.
Isham, Lincoln & Beale
Suite 840
1120 Connecticut Avenue, N.W.
Washington, D. C. 20036

Douglass Cassel, Esq.
109 N. Dearborn Street
Suite 1300
Chicago, Illinois 60602

Ms. Pat Morrison
5568 Thunderidge Drive
Rockford, Illinois 61107

Ms. Lorraine Creek
Rt. 1, Box 182
Manteno, Illinois 60950

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 2
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 September 27, 1985, 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 regulation 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. 2, 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

/S/

Vincent S. Noonan, Director
PWR Project Directorate #5
Division of PWR Licensing-A

Attachment:
Changes to the Technical Specifications

Date of Issuance: FEB 13 1986

No legal objection

OFC	:PD#5	:PD#5	:OELD	:PD#5	:	: OELD	:
NAME	:LOlshan	:MRushbrook	:	:VSNoonan	:VBenaroya	:SH Lewis	:
DATE	:1/7/86	:1/7/86	:/ /86	:1/22/86	:1/17/86	:~~1/27/86~~ 2/7/86	:

OFFICIAL RECORD COPY

ATTACHMENT TO LICENSE AMENDMENT NO. 2

FACILITY OPERATING LICENSE NO. NPF-37

DOCKET NO. STN 50-454

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf page (*) has been provided to maintain document completeness.

REMOVE	INSERT
3/4 3-15	3/4 3-15 3/4 3-16*
3/4 6-7	3/4 6-7 3/4 6-8*
3/4 7-44	3/4 7-44 3/4 7-43*
3/4 8-5	3/4 8-5 3/4 8-6*
3/4 11-8	3/4 11-8 3/4 11-7*
3/4 6-13	3/4 6-13 3/4 6-14*

ONSITE (Continued)

3) Review of all proposed changes to the Technical Specifications;

4) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety;

5) Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Division Vice President and General Manager - Nuclear Stations and to the Supervisor of the Offsite Review and Investigative Function;

6) Review of all REPORTABLE EVENTS;

7) Performance of special reviews and investigations and reports thereon as requested by the Supervisor of the Offsite Review and Investigative Function;

8) Review of the Station Security Plan and implementing procedures and submittal of recommended changes to the Division Vice President and General Manager - Nuclear Stations;

9) Review of the Emergency Plan and station implementing procedures and submittal of recommended changes to the Division Vice President and General Manager - Nuclear Stations;

10) Review of Unit operations to detect potential hazards to nuclear safety;

11) Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Division Vice President and General Manager - Nuclear Stations and the Supervisor of the Offsite Review and Investigative Function; and

12) Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL, and the Radwaste Treatment Systems.

c. Authority

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

ONSITE (Continued)

of action that is more conservative regarding safe operation of the facility. All such disagreements shall be reported immediately to the Division Vice President and General Manager - Nuclear Stations and the Supervisor of the Offsite Review and Investigative Function.

d. Records

1) Reports, reviews, investigations, and recommendations shall be documented with copies to the Division Vice President and General Manager - Nuclear Stations, the Supervisor of the Offsite Review and Investigative Function, the Station Superintendent, and the Manager of Quality Assurance.

2) Copies of all records and documentation shall be kept on file at the station.

e. Procedures

Written administrative procedures shall be prepared and maintained for conduct of the Onsite Review and Investigative Function. These procedures shall include the following:

1) Content and method of submission and presentation to the Station Superintendent, Division Vice President and General Manager - Nuclear Stations, and the Supervisor of the Offsite Review and Investigative Function,

2) Use of committees,

3) Review and approval,

4) Detailed listing of items to be reviewed,

5) Procedures for administration of the quality control activities, and

6) Assignment of responsibilities.

f. Personnel

1) The personnel performing the Onsite Review and Investigative Function, in addition to the Station Superintendent, shall consist of persons having expertise in:

a) Nuclear power plant technology,
b) Reactor operations,
c) Reactor engineering,
d) Chemistry
e) Radiological controls,
f) Instrumentation and control, and
g) Mechanical and electrical systems.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generators, Containment Cooling Fans, Control Room Isolation, Phase "A" Isolation, Turbine Trip, Auxiliary Feedwater, Containment Vent Isolation, and Essential Service Water).					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High-1	3	2	2	1, 2, 3	15*
d. Pressurizer Pressure-Low (Above P-11)	4	2	3	1, 2, 3#	19*
e. Steam Line Pressure-Low (Above P-11)	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3#	15*
2. Containment Spray					
a. Manual Initiation	2 pair	1 pair	2 pair	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High-3	4	2	3	1, 2, 3	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
b. Phase "B" Isolation					
1) Manual Initiation	2 pair	1 pair	2 pair	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure-High-3	4	2	3	1, 2, 3	16
c. Containment Vent Isolation					
1) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	17
2) Manual Phase "A" Isolation	See Item 3.a.1 for all manual Phase "A" Isolation initiating functions and requirements.				
3) Manual Phase "B" Isolation	See Item 3.b.1 for all manual Phase "B" Isolation initiating functions and requirements.				
4) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

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

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures of the running fans at the following locations and shall be determined at least once per 24 hours:

Location

A RCFC Dry Bulb Inlet Temperature

B RCFC Dry Bulb Inlet Temperature

C RCFC Dry Bulb Inlet Temperature

D RCFC Dry Bulb Inlet Temperature

CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specifications 4.6.1.6.1, 4.6.1.6.2, and 4.6.1.6.3.

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

ACTION:

a. With more than one tendon with an observed lift-off force between the predicted lower limit and 90% of the predicted lower limit or with one tendon below 90% of the predicted lower limit, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any abnormal degradation of the structural integrity other than ACTION a. at a level below the acceptance criteria of Specifications 4.6.1.6.1, 4.6.1.6.2, and 4.6.1.6.3, restore the containment vessel to the required level of integrity within 72 hours and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Vessel Tendons. The containment vessel tendons' structural integrity shall be demonstrated at the end of 1, 3, and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. The tendons' structural integrity shall be demonstrated by:

a. Determining that a random but representative sample of at least 19 tendons (5 dome, 6 vertical, and 8 hoop) each have an observed lift-off force within predicted limits for each. For each subsequent inspection one tendon from each group may be kept unchanged to develop a history and to correlate the observed data. If the observed lift-off force of any one tendon in the original sample population lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of this tendon should be checked for their lift-off forces. If both of these adjacent tendons are found to be within their predicted limits, all three tendons should be restored to the required level of integrity. This single deficiency may be considered unique and acceptable. Unless there is abnormal degradation of the containment vessel during the first three inspections, the sample population for subsequent inspections shall include at least 10 tendons (3 dome, 3 vertical, and 4 hoop);

TABLE 3.7-5b (Continued)
(Unit 2)
FIRE HOSE STATIONS

LOCATION		ELEVATION	HOSE RACK REEL	ANGLE VALVE
Aux. Bldg (Continued)				
M-26:	Radwaste Control Panel	387	109	0FP377
M-18:	By Aux. Feedwater motor driven pump 1A	387	108	0FP383
N-23:	By remote shutdown panel U-1	387	111	0FP376
Q-15:	By 480V MCC 132X3	387	113	0FP382
V-18:	By letdown heat exchanger	387	114	0FP379
P-29:	West wall 6.9kV SWGR Rm U-2	455	29	0FP339
L-11:	In UC HVAC Rm OA of LCSR C-1	455	22	0FP332
L-25:	By OBVC HVAC Room	455	27	0FP335
M-8:	South wall of battery room	451	279	0FP638
M-26:	South wall of battery room	451	280	0FP639
M-18:	North wall U-1 AB by door	444	238*	0FP463
L-7:	East wall LCSR A-1	443	207*	0FP330
M-10:	In the southeast corner of LCSR B-1	443	208*	0FP327
P-10:	In the southeast corner of LCSR B-1	443	209*	0FP325
M-13:	South wall of LCSR C-1	443	210*	0FP326
P-13:	West wall of LCSR D-1	443	211*	0FP328
M-18:	South wall of LCSR U-2 AB By Door	443	239*	0FP464
L-29:	East wall LCSR A-2	443	212*	0FP336
M-26:	In Northeast corner of LCSR B-2	443	213*	0FP337
P-26:	In Northwest corner of LCSR B-2	443	214*	0FP340
M-23:	North wall of LCSR C-2	443	215*	0FP341
P-21:	East wall of LCSR D-2	443	216*	0FP333
S-21:	By cabinet 2RY01EC (elec. pen. area)	431	229	0FP454
S-24:	By U-2 cont. shield wall (elec pen. area)	431	230	0FP455
S-15:	By Pzr htr. transformer (elec pen area)	431	236	0FP461
S-12:	By U-1 cont shield wall (elec pen area)	431	237	0FP462
M-18:	Rad Chem Offices	430	57	0FP323
P-11:	Outside Laundry Room	430	52	0FP313
Q-19:	By U-2 VCT valve aisle	430	54	0FP342
P-24:	By radwaste evaporator	430	55	0FP343
V-17:	By east door to decon/change area	430	58	0FP319
V-17:	By west door to decon/change area	430	61	0FP320

*Fire hoses that <u>do not</u> supply the primary means of fire suppression.

TABLE 3.7-5b (Continued)
(Unit 2)
FIRE HOSE STATIONS

LOCATION		ELEVATION	HOSE RACK REEL	ANGLE VALVE
Aux. Bldg. (Continued)				
V-19:	By West Door to Decon Pad & Storage	430	59	0FP344
Q-26:	Back of Div. 21 SWGR Room	430	284	0FP641
S-21:	By U-2 Pzr. HTR. Transformer (elec. pen. area)	419	175	0FP347
Q-26:	By U-2 Elect. Penetration Area	419	206	0FP346
L-11:	By waste oil tank room	405	90	0FP315
P-18:	By elevator	405	91	0FP318
P-23:	By spent resin pumps	405	92	0FP349
Q-11:	By laundry tanks	405	93	0FP314
S-21:	East of U-2 hydrogen recombiner	405	94	0FP348
V-21:	West of U-2 hydrogen recombiner	405	95	0FP345
V-15:	West of U-1 hydrogen recombiner control panel	405	96	0FP316
S-15:	East of U-1 hydrogen recombiner	405	97	0FP317
N-11:	By the recycle holdup tanks	368	130	0FP373
M-13:	By the U-1 stirs	368	131	0FP374
P-13:	By panel 1PL84JB	368	132	0FP369
L-20:	By the U-2 stairs	368	133	0FP355
P-21:	By the blowdown condenser	368	134	0FP356
L-25:	By the PW M/U pumps	368	135	0FP361
N-25:	By chemical drain tank	368	136	0FP357
S-18:	By panel 1PL86J	368	138	0FP362
Q-11:	By Aux. Bldg. floor drain tanks	368	139	0FP368
Q-25:	By spent resin flushing pump	368	137	0FP360
U-21:	By U-2 spray add tank	368	142	0FP358
V-18:	By U-2 cent. chg. pump room	368	141	0FP366
P-11:	By recycle evaporator feed pumps	350	151	0FP381
M-13:	By U-1 stairs	350	152	0FP370
N-23:	By gas decay tanks	350	154	0FP352
Q-19:	By "B" Aux. Bldg. Equip. drain tank	350	155	0FP365
Q-17:	By "A" Aux. Bldg. Equip. drain tank	350	156	0FP371

3) Verifying the diesel generator capability to reject a load of 5500 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection,

4) Simulating a loss of ESF bus voltage by itself, and:

   a) Verifying de-energization of the ESF busses and load shedding from the ESF busses, and

   b) Verifying the diesel starts on the auto-start signal, energizes the ESF busses with permanently connected loads within 10 seconds, energizes the auto-connected safe shutdown loads through the load sequencing timer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the ESF busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on an ESF Actuation test signal without loss of ESF bus voltages, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the generator steady state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss of ESF bus voltage in conjunction with an ESF Actuation test signal, and

   a) Verifying deenergization of the ESF busses and load shedding from the ESF busses;

   b) Verifying the diesel starts on the auto-start signal, energizes the ESF busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through the LOCA sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with emergency loads. After energization, the steady-state voltage and frequency of the ESF busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test; and

   c) Verifying that all automatic diesel generator trips, except engine overspeed and generator differential, are automatically bypassed upon loss-of-voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

7) Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator loading shall be equivalent to the 2-hour rating of 6050 kW* and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 5500 kW. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2f.6)b);**

8) Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 5935 kW;

9) Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

10) Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation and (2) automatically energizing the emergency loads with offsite power;

11) Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines;

12) Verifying that the automatic LOCA and Shutdown sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval; and

---

*Instantaneous loads of 6050 kW (+0, -150) are acceptable as equivalent to the 2-hour rating provided voltage and frequency requirements and cooling system functioning requirements are verified to be within design limits at 6050 kW.

**If Specification 4.8.1.1.2f.6)b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at 5500 kW for 1 hour or until operating temperature has stabilized.

RADIOACTIVE EFFLUENTS

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The Liquid Radwaste Treatment System shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-1) would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31-day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the Liquid Radwaste Treatment System not in operation, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when the Liquid Radwaste Treatment System is not being fully utilized.

4.11.1.3.2 The installed Liquid Radwaste Treatment System shall be considered OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

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.

ISSUANCE OF AMENDMENT NO. 2 TO BYRON UNIT 1 TECHNICAL SPECIFICATIONS - NPF-37

DISTRIBUTION:
Docket File 50-454 *
NRC PDR*
LPDR*
PRC System*
NSIC*
PWR#5 Reading
M. Rushbrook (5)*
L. Olshan*
T.Novak*
S. Lewis, OELD*
OELD*
CMiles, PA
HDenton
R. Diggs, LFMB
JPartlow*
B.Grimes*
EJordan*
L. Harmon*
V. Benaroya*
R. Perch*
TBarnhart (4)*
I.Bailey

*With Tech. Specs.