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August 28, 1981

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U.S. NUCLEAR REGULATORY COMMISSION

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Byron Station Units 1 and 2
Braidwood Station Units 1 and 2
FSAR Revisions
NRC Docket Nos. 50-454, 50-455,
50-456 and 50-457

References (a): July 8, 1981, letter from
R. L. Tedesco to J. S. Abel.

(b): October 31, 1980, letter from
D. C. Eisenhut to All OL and CP Holders.

Dear Mr. Denton:

This is to provide advance copies of answers to questions from the NRC staff regarding the Byron/Braidwood FSAR.

Attachment A to this letter contains responses to questions from the Radiological Assessment Branch transmitted in reference (a). Some voluntary changes to various sections of the FSAR and answers provided previously are also included.

Attachment B to this letter addresses some of the NRC's Post-TMI requirements contained in NUREG-0737 which was transmitted with reference (b). It identifies the requirements which do not apply to Byron and Braidwood. A list of the remaining requirements is also included. They will be addressed by September 18, 1981.

All of this material will be incorporated into the Byron/Braidwood FSAR in the next amendment. Fifteen (15) copies are provided now for your early review and approval. One (1) signed original and fifty-nine (59) copies of this letter are provided.

B001
S 1/1

Please address questions regarding these matters to me.

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator

TRT/lm

Attachments

2466N

List of TMI Action Plan Requirements
Addressed as of August 17, 1981
for Byron and Braidwood Stations
(Appendix E of the B/B FSAR)

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List of TMI Action Plan Requirements
to be Addressed Later

I.A.1.2	I.D.2	II.K.3.9
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I.B.1.2	II.E.1.2	II.K.3.25
I.C.2	II.F.1	II.K.3.30
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I.D.1	II.K.3.7	III.D.3.4

LIST OF TMI ACTION PLAN REQUIREMENTS

WHICH ARE NOT APPLICABLE TO BYRON/BRAIDWOOD STATIONS

IE Bulletins:

II.K.1.20 Prompt Manual Reactor Trip
II.K.1.21 Auto SG Anticipatory Reactor Trip
II.K.1.22 Auxiliary Heat Removal System, Procedures
II.K.1.23 RV Level, Procedures

Orders on B&W Plants:

II.K.2.2 Procedures to Control AFW Independent of ICS
II.K.2.9 FMEA on ICS
II.K.2.10 Safety-Grade Anticipatory Trip
II.K.2.13 Thermal-Mechanical Report
II.K.2.14 Lift Frequency of PORV and SVs
II.K.2.15 Effects of Slug Flow on OTSGS
II.K.2.16 RCP Seal Damage

Final Recommendations, B&O Task Force:

II.K.3.13 HPCI and RCIC Initial Levels
II.K.3.15 Isolation of HPCI and RCIC
II.K.3.16 Challenges to and Failure of Relief Valves
II.K.3.18 ADS Actuation
II.K.3.21 Restart of LPCS and LCPI
II.K.3.22 RCIC Suction
II.K.3.24 Space Cooling for HPCI/RCIC, Modifications
II.K.3.27 Common Reference Level
II.K.3.28 Qualification of ADS Accumulators
II.K.3.44 Evaluate Transients With Single Failure
II.K.3.45 Manual Depressurization
II.K.3.46 Michelson Concerns

E.1 SHIFT TECHNICAL ADVISOR (I.A.1.1)

POSITION:

A technical graduate, licensed at the senior reactor operator (SRO) level, will be provided on each shift at all times when a nuclear unit is in power operation, startup, hot shutdown or cold shutdown (Modes 1-4). A person in this shift position is termed a Station Control Room Engineer (SCRE). The SCRE, functioning as the control room SRO, will also be qualified as a Shift Technical Advisor (STA). The intent is to allow the SCRE to be replaced in the control room by any other SRO on shift as long as the SCRE is within 10 minutes of the control room. This will allow the SCRE the flexibility for in-plant observations from time to time although the SCRE will function, for the most part, in the control room. In this regard the SCRE program is intended to fulfill the long term STA and on shift SRO requirements.

The selection criteria to be used in identifying future SCRE candidates are discussed in Table E.1-1. These criteria are developed in a way that the educational and management qualifications of each prospective candidate are evaluated in light of the guidance, provided by the NRC, for acceptance in the SCRE program. Minor deficiencies identified during this screening process will not disqualify a candidate if it is clear that training provided in the SCRE program will resolve those deficiencies. Other factors not specifically addressed in the NRC criteria such as naval reactors training or other nuclear industry experience, may also contribute to a prospective candidate's qualifications.

The long-term training to be implemented for the SCRE program is discussed in Table E.1-2, which includes a detailed comparison between the Commonwealth Edison SCRE Program and the program outline proposed by the Institute of Nuclear Power Operations (INPO). Also discussed in this table is the SCRE requalification program, which assures the maintenance of technical proficiency of all candidates once they have demonstrated their qualification to go on shift.

The minimum shift manning planned for the two unit, single control room plants is three SROs (a Shift Supervisor, a SCRE [STA], and a Shift Foreman).

TABLE E.1-1

SCRE SELECTION CRITERIA

The SCPE Program is based upon certain trainee prerequisites. These are necessary both as job prerequisites and as background to assure success in training. The limited rate at which SCREs will be qualified makes rigid, prescribed selection criteria unnecessary and undesirable, since there is adequate opportunity for case-by-case consideration. There is still the need, however, to lay out general guidelines by which the case-by-case selection process is conducted.

SCRE SELECTION GUIDELINES

1. The candidate must possess a technical degree; that is, a degree in an engineering or science field. Examples of acceptable fields are: Biology, Chemistry, Computer Science, Environmental Science, Mathematics, Physics, Chemical Engineering, Civil Engineering, Electrical Engineering, Mechanical Engineering, Nuclear Engineering.

2. For candidates with degrees other than Mechanical Engineering or Nuclear Engineering, a careful study of courses taken will be made to identify demonstrated competence in college level mathematics, physics, and chemistry. To identify such competence, consideration will be given to the grade received and the reputation of the college or university for strength in the scientific or engineering field.

3. For candidates with considerable company experience, evaluation of work performed or training completed at the company may be utilized in conjunction with or in lieu of evaluation of past academic performance to identify demonstrated competence in college level mathematics, physics, or chemistry.

TABLE E.1-2

SCRE TRAINING PROGRAM

The following outline describes the SCRE post-SRO Training Program, which is designed to complete the training of an individual who has a technical degree and has completed Senior Reactor Operator training. Depending on the type of technical degree held, some individuals may not require some of the modules in the engineering area, and the requirement for such modules will be waived.

The first modules developed in this program were those to enable the trainees to satisfy the NRC STA training requirements. Development of the remainder of the program is continuing at present, and modules will be administered as they are developed to on-shift SCREs through the requalification program. When all modules have been fully developed, and appropriate long-term scheduling arrangements can be secured with the universities involved, the program will be put into effect, whereby the modules of the SCRE Post-SRO Training Program will be integrated into the SRO training. The purpose will be to develop an individual with a technical degree into a fully-trained SCRE.

Consideration is also being given to individuals who are working toward a technical degree. We intend to seek accreditation for parts of the SCRE pipeline program so that such individuals will be able to complete their degree requirements while in the program. At the time that we institute such a plan, we will revise the SCRE selection guidelines accordingly.

E.3 SHIFT MANNING (I.A.1.3)

POSITION:

Minimum Shift manning will consist of the following:

a. One shift engineer (shift supervisor) with a senior reactor operator's (SRO) license, on site at all times when either Unit 1 or Unit 2 is loaded with fuel.

b. A licensed senior reactor operator (SRO) in the control room at all times when Unit 1 or 2 reactor is in power operation, startup, hot shutdown or cold shutdown (Conditions 1, 2, 3, and 4). The licensed senior reactor operator assigned to the control room may, from time to time, be relieved by the shift engineer (item a, above) or by another licensed senior reactor operator.

c. A licensed reactor operator (RO) in the control room at all times for each reactor containing fuel.

d. An additional reactor operator (RO) on site at all times and available to serve as relief operator for the control room, when either reactor is operating.

Note: A minimum of two reactor operators (ROs) will be provided on the minimum shift crew for one unit operation, and a minimum of three reactor operators (ROs) will be provided on the minimum shift crew for two unit operation.

e. During core alterations, an additional licensed senior reactor operator (SRO) or limited senior reactor operator (SROL) to directly supervise the core alterations. The SRO or SROL may have fuel handling duties but shall not have other concurrent operational duties.

f. Two non-licensed auxiliary operators (AOs) for one unit operation, three AOs for two unit operation.

g. A SCRE/STA on each shift.

Staffing Plan

Five individuals are required for each shift position, in order to provide 24 hr/day, 7 day/week coverage, as well as time for vacation, sickness, and requalification training.

Schedules may vary between different shift positions, but in all cases, the schedules are based on a 40-hour work week. Shifts are normally 8 hour duration (excluding shift turnover time).

Overtime and Work Hours

It is station policy to maintain an adequate number of personnel on the station payroll in the Shift Engineer, Shift Foreman, Station Control Room Engineer, and Nuclear Station Operator job classification such that the use of overtime is not routinely required to compensate for inadequate staffing. Administrative procedures will document the policy concerning this work. Those administrative procedures will reiterate Commonwealth Edison's long standing policy that overtime not be routinely required.

The administrative procedures will also stipulate that work schedules for the Shift Engineer, Shift Foreman, Station Control Room Engineer and Nuclear Station Operator shall be established in advance to ensure that the potential for exceeding the following guidelines is minimized when filling the minimum shift manning requirements previously defined; that is:

a. No individual should work more than 12 consecutive hours. This does not include time necessary for shift turnover.

b. No individual should work more than 24 hours in any 48 hour period.

c. No individual should work more than 72 hours in any 7 day period.

d. No individual should work more than 14 consecutive days without having 2 consecutive days off.

It is understood that there may be short-term requirements for exceeding the above guidelines due to unexpected illness, unusual Unit conditions, etc. Vacancies shall be covered in accordance with the overtime rules of the Collective Bargaining Agreement and existing Station guidelines. Those instances which require deviation from the above overtime limitation guidelines shall be documented and reviewed by the Station Superintendent as soon as practicable following the occurrence.

E.9 GUIDANCE FOR THE EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS (I.C.1)

POSITION:

As stated in References 1 and 2, this item is still in progress. Our procedures will be developed when the Westinghouse Owners' Group have completed their effort.

E.12 PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF (I.C.5)

POSITION:

Commonwealth Edison has in-place corporate procedures to direct feedback of operating experience at operating stations under guidance of the Director of Nuclear Safety. The Byron and Braidwood Stations are part of this feedback mechanism at the present time even though they are not operating stations. We therefore consider this item complete.

E.19 REACTOR COOLANT SYSTEM VENTS (II.B.1)

POSITION:

The reactor coolant system vent (RCSV) line is located at the top of the reactor integrated head. This 0.5 inch diameter schedule 160 line contains four safety grade solenoid-operated valves which are powered by emergency buses. Being located at a high point permits this line to vent the reactor coolant system normally connected to the reactor pressure vessel. The RCSV is remotely operated and monitored from the main control room. Since the RCSV line is a 0.5 inch pipe, it is smaller than the size for which a LOCA analysis would be required.

The RCSV line was designed and installed as ASME Section III, Class 1 piping to applicable codes. Final positioning of the discharge of the RCSV minimizes possible impingement on equipment or obstructions. (See Figure E.19-1, RCSV ISOMETRIC DRAWING.)

Seismic and environmentally (IEEE 323-1974) qualified ASME Section III Class 1 solenoid-operated valves (1(2) RC014A-D) are installed in parallel sets of two, supplied by redundant emergency buses. Positive indication of valve position is provided, from valve operator limit switches, to the control switch lights in the main control room.

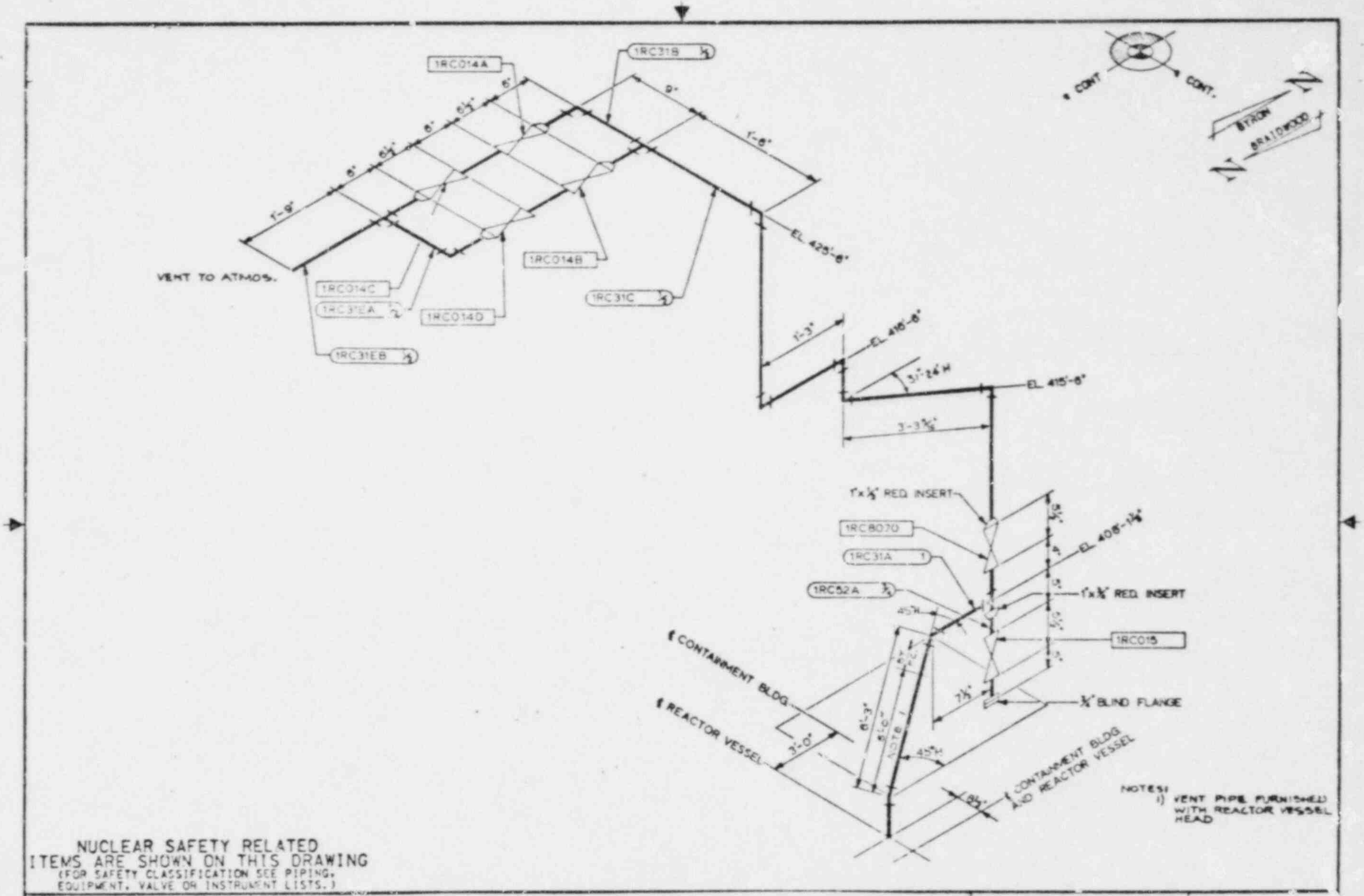
A main control room alarm is also provided in conjunction with valve position indication to alarm when any vent valve is open. In addition, surface mounted resistance temperature detectors with main control room alarms are provided downstream of the solenoid-operated valves for leak detection.

These valves are designed to pass steam, steam/water, water, and non-condensible gases. The RCS vents directly to the containment. Possible hydrogen concentration will be controlled by the containment hydrogen recombiners.

The Westinghouse Owners' Group, of which the Commonwealth Edison Company is a member, is working on guidelines and procedures for use of the RCSV system.

Complete analysis of the RCSV system is not yet completed.

Human factor analysis will be taken into account in finalizing the Byron and Braidwood Stations emergency procedures and monitoring equipment with respect to the use of the reactor coolant vent system.



REACTOR VESSEL HEAD
VENT SYSTEM
CONTAINMENT BUILDING

BYRON/BRAIDWOOD UNIT 1
COMMONWEALTH EDISON CO.
CHICAGO, ILLINOIS

FIGURE E.19-1

E.20 PLANT SHIELDING (II.B.2)

POSITION:

A radiation and shielding design review was conducted for Byron/Braidwood Stations (B/B) using the guidance provided in NUREG-0737. Source terms were developed based on a release of 100% of the noble gases, 50% of the halogens, and 1% of all remaining fission products from the fuel. The post-accident radiation environment was determined by (1) analyzing each system operating following an accident to establish pathways for fission products out of containment; (2) investigating process streams in the auxiliary building to identify contaminated equipment and associated activity levels; (3) calculating the radiation field due to each source with computer codes ISOSHLD (Ref. 3), QAD (Ref. 4), and G^3 (Ref. 5); and, (4) superimposing the effects of all sources to obtain the maximum expected dose rate throughout the plant. The radiation environment was evaluated 1 hour, 1 day, and 1 week following the reactor shutdown that precedes fuel failure.

The potentially contaminated systems considered in identifying postaccident radiation sources included (1) the emergency core cooling system, which consists of all or parts of the safety injection, residual heat removal, and chemical and volume control systems; (2) the containment spray and hydrogen recombiner systems, which assist in maintaining containment integrity; (3) the control room, technical support center, and auxiliary building HVAC systems, portions of which are designed to remove airborne fission products; (4) the high radiation sampling system; (5) those parts of the chemical and volume control and the boron recycle systems associated with charging, letdown, and seal water; (6) the shutdown cooling portion of the residual heat removal system; and, (7) those parts of the gaseous waste processing system and the liquid radwaste system which would normally operate in conjunction with the other systems under consideration. The large amount of equipment in the auxiliary building operating following an accident produces elevated dose rates in many areas. However, since no operator actions other than those which take place in the control room or at the remote shutdown panel are critical for plant shutdown, only these areas along with the technical support center, the sampling station, and the hydrogen recombiner control panels are considered to be vital for personnel access for postaccident operations.

In genera', the shielding design review shows that personnel occupancy in the vital areas will not be unduly restricted by postaccident radiation fields. A significant radiological conclusion is that the "less than 15 mr/hr" criterion is met at B/B for plant areas requiring extended or continuous occupancy (the control room, the technical support center, and the radwaste control room, where the remote shutdown panels are located). Additionally, application of General Design Criterion 19 accident limit of 5 rem whole body (or equivalent) for areas requiring infrequent access indicates that adequate occupancy times are available for typical operator actions in the remaining vital areas. Postaccidnet dose rates from contained sources are shown in Table E.20-1 for each of the vital areas. Table E.20-2 gives the post-accident doses for personnel transit between selected plant control centers and other postaccident vital areas. This table gives a conservative estimate of the travel dose due to contained sources in the auxiliary building and airborne activity in containment. The location of postaccident vital areas and areas essential for access to vital areas (stairways) are indicated in Figure E.20-1.

The evaluation of the environmental qualifications of essential equipment, which demonstrates that the equipment will not be unduly degraded by postaccident radiation fields, will be provided later.

TABLE E.20-1

POSTACCIDENT DOSE RATES FROM CONTAINED SOURCES

(in rem/hr)

LOCATION	1-HOUR	1-DAY	1-WEEK
Control Room	<0.001	<0.001	<0.001
Remote Shutdown Panel	<0.015	<0.015	<0.015
Technical Support Center	<0.001	<0.001	<0.001
Primary Sample Room	<0.1	<0.1	<0.1
Laboratories	<0.015	<0.015	<0.015
Hydrogen Recombiner Control Panel			
Unit 1	<0.1	<0.1	<0.1
Unit 2	<0.015	<0.015	<0.015
Pathway to TSC	<1.0	<0.015	<0.015
Pathway to Remote Shutdown Panel	<0.015	<0.015	<0.015

TABLE E.20-2

POSTACCIDENT DOSES FOR ESSENTIAL POSTACCIDENT PATHS[3]

(DOSES ARE IN MILLIREMS FOR SELECTED POSTACCIDENT TIMES)

PATHWAY	WALK[1]			RUN[2]		
	1 Hour	1 Day	1 Week	1 Hour	1 Day	1 Week
TSC to CR	5.77	0.33	0.02	2.89	0.17	0.01
RSCP to CR	0.42	0.42	0.03	0.25	0.25	0.02
LABS to CR	0.02	0.02	(4)	0.01	0.01	(4)
HRSS to LABS	1.39	0.21	0.01	0.80	0.12	0.01
HRCP-1 to CR	0.29	0.29	0.02	0.16	0.16	0.01
HRCP-2 to CR	0.01	0.01	(4)	0.01	0.01	(4)

CR - Control Room, TSC - Technical Support Center, RSCP - Remote Shutdown Control Panel, LABS - Laboratories, HRSS - High Radiation Sample System, HRCP-1 (2) - Hydrogen Recombiner Control Panel, Unit 1 (Unit 2).

NOTES:

1. Walking Speeds: 300 ft/min horizontal, 50 ft/min upstairs, 90 ft/min downstairs.
2. Running Speeds: 600 ft/min horizontal, 80 ft/min upstairs, 120 ft/min downstairs.
3. Sources: All contained sources in the auxiliary building and airborne actiivty in containment based on NUREG-0737. Dose due to plume is not included.
4. Dose is less than 0.01 millirems.

FIGURE E.20-1
(LATER)

E.21 POSTACCIDENT SAMPLING (II.B.3)

POSITION:

The capability to obtain and perform radioisotopic and chemical analyses of the reactor coolant and containment atmosphere samples is provided by the high radiation sampling system (HRSS), the design of which is outlined in the following paragraphs. The system will be installed and become operational prior to full power operation.

GENERAL SYSTEM DESCRIPTION:

The P&ID of the HRSS for Unit 1 is shown in Figure 9.3-3, Sheets 1 through 3. The system is installed in the auxiliary building and consists of a liquid sampling subsystem and an air sampling subsystem. The major components of the system are:

a. HRSS liquid sample panel,

b. liquid sample cooler rack,

c. chemical analysis panel,

d. chemical analysis monitor panel,

e. HRSS auxiliaries control panel (liquid subsystem only),

f. waste drain tank and pumps,

g. containment air sample panel (CASP),

h. CASP control panel, and

i. valves and piping for the system.

The liquid sampling subsystem is installed at elevation 401 feet in the auxiliary building except for the waste drain tank and pumps which are installed at elevation 383 feet. The air sampling subsystem is installed in the auxiliary building in proximity to the containment.

LIQUID SAMPLING SUBSYSTEM

The HRSS liquid sampling panel is capable of sampling:

a. pressurizer steam space,

b. pressurizer liquid space,

c. reactor coolant hot leg loops 1 and 3,

d. reactor coolant cold leg loops 1 through 4,

e. RHR heat exchangers A and B outlets,

f. reactor coolant letdown heat exchanger outlet,

g. CVCS demineralizer outlet,

h. BTR demineralizer outlet,

i. reactor coolant filter outlet,

j. auxiliary building floor drain tank A (B on Unit 2 panel),

k. auxiliary building equipment drain tank A (B on Unit 2 panel),

l. recycle holdup tank A (B on Unit 2 panel),

m. HRSS waste drain tank,

n. containment floor drain sump,

o. chemical drain tank (Unit 1 panel only),

p. steam generator blowdown sample line, and,

q. regeneration waste drain tank (Unit 1 panel only).

In addition to taking the above samples for onsite and/or offsite analysis, the HRSS liquid sampling panel is capable of routing the reactor coolant samples to the chemical analysis panel. The chemical analysis panel is capable of performing the on-line analysis of pH, dissolved oxygen, specific conductivity, chloride, and hydrogen. For boron and isotopic analysis, samples diluted by a factor of 1000 to one will be transferred to the onsite laboratory. Excessive exposure to the system operator is limited by:

a. lead shielding in the liquid sampling panel and the chemical analysis panel;

b. concrete shielding above, below, and around the sides of the panels to prevent radiation from scattering around the lead shielding;

c. the optimized design and reduced amount of piping in the panels containing reactor coolant;

d. a special cart equipped with a shielding cask to transport the radioactive sample to its destination; and,

e. a ventilation system drawing air out of the sampling panels and discharging into a remote HVAC train.

AIR SAMPLING SUBSYSTEM

The containment air sampling panel is capable of sampling the primary containment atmosphere. The sample is drawn from the containment through a dedicated penetration. Once the interfacing valves are arranged and the sampling programmer is inititiated, the containment air sampling panel utilizes automatically sequenced sampling to trap the designated sample in a shielded cart. The air sample will then be analyzed onsite. Excessive exposure to the operator is limited by:

a. steel shielding in the containment air sampling panel;

b. concrete shielding above, below, and around the sides of the panel to prevent radiation from scattering around the steel shielding;

c. automatic sampling;

d. special carts each equipped with a shielding cask to transport the radioactive sample to its destination; and,

e. a ventilation system drawing air out of the sampling panels and discharging into a remote HVAC train.

SAMPLING PROGRAM FREQUENCY

Actual frequency of sampling shall be determined by station management, however, as a minimum the first sample can be taken within 1 hour from the time a decision is made to take a sample, continuing with at least one sample per day for the next 7 days and at least one sample per week thereafter. The time interval between taking a sample and receipt by plant management of the results of the analysis is estimated to be less than 2 hours.

E.23 RELIEF AND SAFETY VALVE TEST REQUIREMENTS (II.D.1)

POSITION:

By letter dated July 1, 1981 (Ref. 4), R. C. Youngdahl (Consumer Power) transmitted the Interim Data Report for the EPRI PWR Safety and Relief Valve Test Program. This report summarizes the test data collected to date on relief valves (safety valve data is still not available). Byron/Braidwood plants have Copes-Vulcan Model D-100-160 2-inch air-operated globe relief valves (316SS w/stellite plug and 17-4PH cage) and Crosby Model HP-BP-86, size 6M6 safety valves. Results provided in Section 4.6 of the EPRI Interim Data Report are applicable to Commonwealth Edison plants. Preliminary evaluation of the data indicates that the relief valves will perform their intended function for steam inlet conditions. Commonwealth Edison will submit preliminary and final evaluations, and other plant specific data for all relief evaluations, and other plant specific data for all relief and safety valve inlet conditions yet to be tested on a schedule consistent with the R. C. Youngdahl letter of December 15, 1980 (Ref. 5) and modified on July 1, 1981.

E.25 AUXILIARY FEEDWATER SYSTEM EVALUATION (II.E.1.1)

POSITION:

As a response to FSAR Question 10.53, the following analysis of the Byron/Braidwood auxiliary feedwater system was done:

1. A point-by-point review of the auxiliary feedwater system design against Subsection 10.4.9 of the Standard Review Plan and Branch Technical Position ASB 10-1;

2. A reliability study discussed in NUREG-0611;

3. A point-by-point review of the auxiliary feedwater system design; technical specifications and operating procedures against the generic short-term and long-term requirements discussed in the March 10, 1980 letter (Ref.8); and,

4. A design basis evaluation of the auxiliary feedwater system.

This effort has been submitted for review. It will be an appendix to the B/B-FSAR. We therefore consider this item complete.

E.27 EMERGENCY POWER SUPPLY FOR PRESSURIZER HEATERS (II.E.3.1)

POSITION:

As outlined in Chapter 8.0 of the FSAR, the Byron/Braidwood distribution system is designed with Class 1E qualified breakers between the 4160-volt ESF buses and the 4160-volt non-safety-related buses. They can be closed to provide emergency power to the pressurizer heaters by manual operator action. For Byron/Braidwood, one bank of backup heaters from each redundant power supply can be connected within 60 minutes to maintain natural circulation after loss of offsite power. The circuits are designed to automatically shed from the emergency power sources upon the occurrence of a safety injection actuation signal. Procedures will be established to manually load the pressurizer heater on to the emergency power sources.

E.28 CONTAINMENT-DEDICATED PENETRATIONS (II.E.4.1)

POSITION:

Two permanently installed hydrogen recombiners are provided at the Byron/Braidwood Stations. The suction, discharge, and cross-tie piping and valves are ASME III, Class 2 and utilize penetrations dedicated for this system. The containment purge system is now safety-related and utilizes dedicated penetrations per code requirements for the purge system. Redundancy and single-failure requirements of General Design Criteria 54 and 56 of 10 CFR 50, Appendix A are met.

E.29 CONTAINMENT ISOLATION DEPENDABILITY (II.E.4.2)

POSITION:

The containment isolation system on Byron/Braidwood, as shown in the functional diagrams, Figure 7.2-1 of the FSAR, is an automatic, redundant, safety grade system which has diverse parameters for actuation. In Table 6.2-58 of the FSAR a summary of containment isolation signal and essential and nonessential systems that provide a possible open path out of the primary containment through Class B penetraton are either automatically isolated by isolation signals, by check valves that would prevent flow out of the containment, by manual valves that are normally closed during reactor operation, or as in the case of instrument lines, by closed piping system. The individual control circuits are designed to prevent automatic loss of containment isolation due to resetting of the isolation signals.

E.32 EMERGENCY POWER FOR PRESSURIZER EQUIPMENT (II.G.1)

POSITION:

The motor-operated pressurizer relief isolation valves and the solenoid air-operated pressurizer power relief valves are qualified Class 1E devices per Reference 1 of FSAR Subsection 3.11.6. Therefore, their motive and control power is supplied from qualified emergency buses at all times. In addition, the source supplying the relief isolation valves is different from the source supplying the power relief valves. There is also the capability to manually connect the instrument air to the emergency buses for the power relief valves if required. The pressurizer level indication on Byron/Braidwood is designed for postaccident monitoring, and therefore, the instrument channels are always powered from the vital instrument buses.

E.33 ASSURANCE OF PROPER ESF FUNCTIONING (II.K.1.5)

POSITION:

Valve positioning requirements, positive controls, and test and maintenance procedures associated with ESF systems have been reviewed. Motor-operated valves in safety systems are normally maintained in a configuration such as to require the least number of valve automatic movements upon system actuation. System initiation logic is such that valves automatically move to the required position when required. The position of vital manual ECCS valves is controlled by the use and documentation of locks on valve handwheels.

Byron/Braidwood Stations are equipped with an ESF status panel which continuously monitors the ESF systems for any deviation which would indicate the system is not available. Typical parameters monitored include:

a. Valve position.

b. Power available to motor-operated valves.

c. Initiation logic power available.

d. Power sources (including emergency diesels) available.

e. Breaker status.

Alarms are provided on a system level if the system is not in a ready mode.

Surveillance and testing procedures for ESF systems include checklists to ensure that individual systems return to ready status upon completion of testing.

When ESF equipment is removed from service for maintenance, the Commonwealth Edison Equipment Outage Procedure requires documentation of removal and return to service. Functional tests of equipment returned to service are required by this procedure to ensure operability.

In accordance with general station practices, all ESF systems are verified to be lined-up in accordance with approved mechanical and electrical checklist prior to fuel load.

E.34 SAFETY-RELATED SYSTEM OPERABILITY STATUS ASSURANCE (II.K.1.10)

POSITION:

During the preoperational test program at Byron/Braidwood (B/B), the emergency power redundancy preoperational test (see FSAR Table 14.2-1) will be conducted to verify the independence and redundancy of each of the two EFS division power supplies. As outlined in Regulatory Guide 1.40, the test will be conducted for both energized and deenergized conditions of the division not under test.

The PWR Standardized Technical Specification and the B/B Technical Specification (see FSAR Chapter 16.0) clearly states that performance of a surveillance requirement within the specified time interval shall constitute compliance with operability requirements for limiting conditions for operations (LCO) for operation and associated Action Statements, unless otherwise required by the specification. This section of the Technical Specification eliminates the requirements of previous Technical Specifications to test redundant components when a component was determined to be inoperable.

The ESF systems at B/B as described in the various chapters of the FSAR are designed to be electrically and mechanically independent and redundant where required. Based on the design redundancy of station systems and the above information, the B/B station procedures require operability testing of redundant components remaining in service as a part of a redundant system only when specifically required by the B/B Technical Specifications.

E.49 INSTALLATION AND TESTING OF AUTOMATIC POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM (II.K.3.1)

POSITION:

See position taken on Clarification Item II.K.3.2, on page E.50-1 of this FSAR.

E.50 REPORT ON OVERALL SAFETY EFFECT OF POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM (II.K.3.2)

POSITION:

The Westinghouse Owners' Group letter #OG-52 dated March 13, 1981 (R. Jurgenson letter to J. R. Miller) was submitted recommending against the installaton of automatic PORV isolation (Item II.K.3.1). Commonwealth Edison concurs with that recommendation. Therefore, in our judgment, no further action is required at this time.

E.52 AUTOMATIC TRIP OF REACTOR COOLANT PUMPS DURING LOSS-OF COOLANT ACCIDENT (II.K.3.5)

POSITION:

Response to this item was made by the Westinghouse Owners' Group in the June 15, 1981 letter from R. W. Jurgensen to P. S. Check. Additional submittals concerning this item, if necessary, will be made in accordance with the schedule provided in the Owners' Group June 15, 1981 letter above.

E.57 CONFIRM EXISTENCE OF ANTICIPATORY REACTOR TRIP UPON TURBINE TRIP (II.K.3.12)

POSITION:

Th reactor protection system contains an anticipatory reactor trip on turbine trip. This trip is described in FSAR Subsection 7.2.1.1.2(f) and is required by Technical Specification 3.3.1.1. Therefore, no modification is proposed or required.

E.77 PRIMARY COOLANT SOURCES OUTSIDE CONTAINMENT (III.D.1.1)

POSITION:

A program is being developed to monitor leakage from systems outside the containment which could be used to transport highly radioactive fluids and gases in a postaccident condition. This program includes the following features:

a. A combination of general inspections and detailed system walkdowns of liquid systems. These inspections are performed with the system operating at approximately expected pressures during normal operation or during tests.

b. Systems containing gases are to be tested by use of tracer gases (DOP, freon or helium), by pressure decay testing, or by metered make-up tests.

c. A maintenance program is used to assign high priorities to radioactive fluids related leakage work requests.

d. A summary description of this program, including a systems list and initial leak test results, will be submitted to the NRC 4 months prior to the issuance of a fuel-loading license.

e. Leakage related work requests are to be reviewed to evaluate possible modifications to keep leakage and exposures "as low as reasonably" possible.

This program will be implemented prior to issuance of a full-power license.

E.80 REFERENCES

1. Letter from B.J. Youngblood (NRC) to J.S. Abel (Commonwealth Edison) dated June 11, 1981, regarding NUREG-0737 item I.C.1.

2. Letter from D.G. Eisenhut (NRC) to R.W. Jurgensen (Commonwealth Edison) dated May 28, 1981.

3. R.L. Engel, J. Greenborg, and M.M. Hendrickson, "ISOSHLD -- A Computer Code for General-Purpose Isotope Shielding Analysis," BNWL-236, Pacific Northwest Laboratory, Richland, Washington, June 1966; Supplement 1, March 1967; Supplement 2, April 1969.

4. R.E. Malenfant, "QAD -- A Series of Point-Kernel General-Purpose Shielding Programs," LA-3573, Los Alamos Scientific Laboratory, April 5, 1967.

5. R.E. Malenfant, "G^3 -- A General-Purpose Gamma-Ray Scattering Program," LA-5176, Los Alamos Scientific Laboratory, June 1973.

6. Letter from R. C. Youngdahl (Consumers Power) to H.R. Denton (NRC) dated July 1, 1981.

7. Letter from R.C. Youngdahl (Consumers Power) to D.G. Eisenhut (NRC) dated December 15, 1980.

8. Letter from D.F. Ross, Jr. (NRC) to all pending operating license applicants of nuclear steam supply systems designed by Westinghouse and Combustion Engineering, dated March 10, 1980.

Byron/Braidwood Stations

Responses to Radiological Assessment

Branch (331 Series) Questions

The following ten questions in the 331.24 - 331.35 series have been responded to:

331.24

331.25

331.26

331.27

331.28

331.29

331.30

331.31

331.32

331.34

Voluntary FSAR changes and revised responses to questions 422.13 and 422.14 are also included.

QUESTION 331.24

"Based on information contained in NUREG-0731 'Criteria for Utility Management and Technical Competence,' it is our position that the radiation protection group should be a separate organizatin from the chemistry group. Your station organization chart (Figure 6.1-2) shows the radiation protection and chemistry technicians combined. Additionally, in accordance with Regulatory Guide 8.8, it is our position that the Rad-Chem Supervisor (RPM) should have access to the Station Superintendent in all radiation protection matters. In matters related to radiological health and safety, the RPM has direct responsibility to both employees and management that can best be fulfilled if he is independent of station divisions, such as operations, maintenance or technical support, whose prime responsibility is continuity or improvement of station operability. Your FSAR should be revised to outline how your planned radiation protection program reflects this position.

"Concurrent to the changes requested above, Figure 6.1-2 should show that on radiation protection matters, the Rad-Chem Supervisor has access to the Station Superintendent and the radiation protection technicians and chemistry technicians become separate groups. In addition the technicians should be qualified separately as chemistry and radiation protection technicians, and each report directly to their respected group supervisor."

RESPONSE

Topical Report CE-1A, Revision 14, dated September 9, 1980 delineates the line of authority from the Rad/Chem Supervisor to the Station Superintendent. This Topical Report was submitted to the Commission on September 18, 1980 and supersedes previous organizational information.

NUREG-0731, "Criteria for Utility Management and Technical Competence," provides a representative plant organization chart. The criteria state that organizational arrangements can vary considerably to include the characteristics of the chart. All Commonwealth Edison nuclear stations currently utilize a common radiation protection/chemistry department as described in Topical Report CE-1A. Byron/Braidwood Stations will conform to the organization as described in Topical Report CE-1A.

QUESTION 331.25

"In accordance with the criteria contained in NUREG-0731, it is our position that your organization chain should contain a qualified individual to provide backup in the event of the absence of the Radiation Protection Manager (RPM). The December 1979 revision of ANSI 3.1 specifies that individuals temporarily filling the RPM position should have a B.S. degree in science or engineering, two years experience in radiation protection, one year of which should be nuclear power plant experience, six months of which should be on-site. It is our position that such experience should be professional experience. Identify and provide an outline of the qualification of the individual who will act as the backup for the RPM in his absence."

RESPONSE

For Byron and Braidwood Stations, individuals temporarily filling the Radiation Protection Manager (RPM) position will have a B.S. degree in science or engineering, two years experience in radiation protection, one year of which should be nuclear power plant experience, six months of which should be onsite. However, in accordance with Section 4.1, paragraph 2 of Qualifications, ANSI 3.1, December 1979 draft, "Individuals who do not possess the formal educational requirements in this section shall not be automatically eliminated where other factors provide sufficient demonstration of their abilities."

QUESTION 331.26

"Table 13.1-2 of your FSAR states that you will fill the position of Rad-Chem Supervisor 45 months prior to core loading. Your Technical Specifications will require your Rad-Chem Supervisor to meet the criteria for Radiation Protection Manager as specified in Regulatory Guide 1.8. You should provide a resume of the education, training, and experience of the individual selected to fill this position as soon as it is available."

RESPONSE

A member of the Radiation/Chemistry Department will meet or exceed the requirements of Regulatory Guide 1.8, Revision 1, for the position of Radiation Protection Manager. This person may be the Rad/Chem Supervisor or a technical individual reporting to the Rad/Chem Supervisor.

A Rad/Chem Supervisor was appointed in June of 1978 at Byron Station, 46 months prior to the currently scheduled core loading of Unit #1. The resume for the Rad/Chem Supervisor has been added to B/B FSAR Attachment 13A.

A Rad/Chem Supervisor was appointed in December of 1978 at Braidwood Station, 76 months prior to the currently scheduled core loading of Unit #1. The resume for the Rad/Chem Supervisor has been added to B/B FSAR Attachment 13A.

QUESTION 331.27

"Section 13.1.3.1 of the FSAR states that 'an individual will be considered qualified as a radiation-chemistry technician in a responsible position after completion of the onsite radiation-chemistry technician training program, and a total of one year of general power plant, chemistry, or radiation protection experience or equivalent training.' Your Technical Specifications will require that your plant staff meet the qualification of ANSI 18.1. This ANSI standard requires two years experience in radiation protection before an individual is considered qualified as a radiation protection technician. If you wish to propose an alternative qualification program for your radiation technicians, you should provide a detailed description of the qualification program. Otherwise, your FSAR should be amended to show radiation technician qualification per ANSI 18.1."

RESPONSE

Refer to revised B/B-FSAR Subsection 13.1.3.1 for a revised qualification program for Radiation Chemistry Technicians.

QUESTION 331.28

"Your portable radiation monitoring instrument list in Table 12.5-2 shows no instrument capable of measuring dose rates greater than 100 R/hr. Instruments capable of measuring 10,000 R/hr are necessary to determine radiation levels following a TMI-type accident. Regulatory Guide 1.97 (Revision 2) specifies that portable survey meters and the area radiation monitors in areas requiring access after an accident should have a range up to 10,000 R/hr. Provide a commitment in your FSAR to have such portable radiation monitoring instruments and specify locations of area radiation monitors in accessible post-accident areas. These monitors should be capable of measuring dose rate up to 10,000 R/hr."

RESPONSE

Table 12.5-2 of the B/B-FSAR states that portable high range dose rate instruments will have a minimum range of 0 to 100 R/hr.

Table 12.5-2 will be revised to provide for instruments with a range of 0 to 1,000 R/hr. Portable instruments with a range from 1,000 R/hr to 10,000 R/hr are not committed to since:

a. Such instruments are not yet commonly available; and

b. It is unlikely that an individual would ever be allowed to enter an area with a suspected radiation field of those magnitudes. Area radiation monitors are installed in areas where high range indication may be necessary.

QUESTION 331.29

"Provide information on the quantity and types of respirators provided for Byron/Braidwood Station in accordance with the guidelines of Regulatory Guide 1.70, Section 12.5.2."

RESPONSE

Respiratory protection programs are established in accordance with Regulatory Guide 8.15, Revision 0.

Section 12.5.. of Regulatory Guide 1.70 requests type and location of respiratory protective equipment. This information is provided in B/B FSAR Table 12.5-1. In addition to the equipment listed in Table 12.5-1 an emergency breathing air system exists for Control Room personnel.

Quantities of respirators . l vary and will be sufficient to meet personnel needs.

QUESTION 331.30

"Verify that your program for internal radiation exposure assessment (whole body counting, and bioassay programs) meets the criteria of Regulatory Guide 8.26, 'Applications of Bioassay of Fission and Activation Products' or outline equivalent alternatives."

RESPONSE

The B/B Internal Exposure Control Program is described in B/B FSAR Subsection 12.5.3.3.

Both direct and indirect bioassays are conducted in order to assess internal radiation exposures. Commonwealth Edison utilizes a contractor to provide bioassay services on a routine and non-routine basis. The contract itself provides for a quality assurance program and adherence to all applicable regulations. The program used at the B/B Stations is identical to that used at other Commonwealth Edison operating nuclear stations.

QUESTION 331.31

"Verify that your air sample calibration program meets Regulatory Guide 8.25 or outline an equivalent alternative."

RESPONSE

The B/B air sample programs comply with guidance outlines in Sections C.1, C.2, and C.3 of Regulatory Guide 8.25, dated August, 1980.

QUESTION 331.32

"Provide additional information on how your exposure tracking and exposure reduction program, includes the elements of Regulatory Guide 1.70, Section 12.1.3 and 12.5.3, and Regulatory Guide 8.8, Section C.3.9(8)(j), C.3.8(2), and C.3.c(2)(5), including rem-tracking, self-reading pocket dosimeter use, post-maintenance, actual exposure, and how these results are used to make changes in future work. Verify that annual exposure reviews are performed by plant management and that these are used to identify groups with the highest exposure in order to assure that doses are ALARA."

RESPONSE

The Commonwealth Edison commitment to the ALARA principle is discussed in B/B FSAR Subsection 12.1.1. The use of Radiation Work Permits is discussed in B/B FSAR Subsection 12.1.1.3.

Pencil dosimeters will be used at Byron/Braidwood Stations to record estimates of daily exposures received by each individual worker. This information enables the Rad/Chem Department to spot significant individual exposures that may occur within the biweekly film badge monitoring period. Biweekly work group man-rem summaries are generated by the computer dosimetry program. The summaries serve to alert the station health physics staff and the corporate office of the trends in man-rem expenditures. Commonwealth Edison began a Radiation Evaluation Program (REP) in April of 1976. REP is a computer based occupational dose accounting system used to document, by work group, the dose expenditure resulting from work performed on various plant systems and components. In addition to each work group's dose and the plant component worked on, the program will document the total work effort in man-hours and include a brief description of the work performed.

The REP program applications are:

a. To provide timely radiological feedback information to our engineering and production departments and architect-engineer consultants for consideration in new plant design and to enable corrective action to be taken at existing stations.

b. To identify and compile dose histories on specific sources of occupational dose that might be reduced through improved station working and shielding procedures and training programs.

c. To provide data for comparison studies of specific sources of occupational exposure among similar CECo nuclear stations with relevant factors such as reactor equipment and plant layout, etc., taken into account.

d. To demonstrate an "active ALARA program."

The Station is also planning for an ALARA Review Committee. This committee is composed of the manager of each affected department, the Rad/Chem Supervisor, and an ALARA coordinator. The charter of the committee is to advise the Station Superintendent on ALARA matters. The committee reviews annual exposure reduction goals and provides direction for the ALARA coordinator. The committee meets at least quarterly. The chairman of the committee has decision making responsibility.

QUESTION 331.34

"Explain how the Byron/Braidwood Station will implement a chemistry control program to reduce radiocobalt production and crud buildup in normally radioactive systems. (See Regulatory Guide 8.8, Section C.2.e(3))."

RESPONSE

Radiocobalt and crud buildup in the primary coolant above 250° F will be controlled below specification limits by continuous monitoring and controlling of the oxygen concentration. Hydrazine additions to the primary coolant and a hydrogen or nitrogen blanket in the volume control tank will be the means of oxygen control. Control of pH in the primary coolant will be accomplished by lithium hydroxide addition and will be maintained between a pH of 4.5 and 10.5 depending on the disassociation of the boric acid present in the primary coolant.

QUESTION 422.13

"Your qualification requirements for the position of Rad/Chem Supervisor is unacceptable. It is the staff's position, in regard to the health physics qualifications, that the qualification reqirements for this should meet those described in Revision 1 to Regulatory Guide 1.8 for the position of Radiation Protection Manager."

RESPONSE

For the Byron and Braidwood Stations, it is Commonwealth Edison's position that a member of the Rad/Chem group will meet the qualifications of the Radiation Protection Manager as defined in Regulatory Guide 1.8, Revision 1. This person may be the Rad/Chem supervisor or a technical individual reporting to the Rad/Chem Supervisor.

QUESTION 422.14

"The qualification requirements for the position of Radiation Chemistry Technicians are not satisfactory. It is our position that the qualifications for this position should meet that described in Section 4.5.2 of ANSI N18.1-1971."

RESPONSE

See the response to Question 331.27.

TABLE 12.5-2

HEALTH PHYSICS EQUIPMENT

TYPE DETECTOR/MONITOR	ESTIMATED NUMBER	SENSITIVITY	RANGE	FREQUENCY	CALIBRATION METHOD
Multichannel Analyzer	2	Variable	Variable	Annual with Quarterly Check	Standard Reference Materials
Alpha/Beta Counting System	2	Variable	Variable	Annual with Quarterly Check	Standard Reference Materials
Air Ion Chamber Dose Rate Meter	20	Variable	Variable	Annual with Quarterly Check	Standard Reference Materials
GM Survey Count Rate Inst.	10	Variable	Variable	Annual with Quarterly Check	Standard Reference Materials
Alpha Scintillator Probe	2	Variable	0-100K cpm Minimum	Annual with Quarterly Check	Standard Reference Materials
Hi Range GM Dose	2	Variable	0-100R/HR Minimum	Annual with Quarterly Check	Standard Reference Materials
Neutron BF_3	2	Variable	0-1R/HR Minimum	Annual with Quarterly Check	Standard Reference Materials/Mini Pulser
Air Sample	10	N/A	Variable	Annual with Quarterly Check	Air Flow Calibrator

TABLE 12.5-2 (Cont'd)

TYPE DETECTOR/MONITOR	ESTIMATED NUMBER	SENSITIVITY	RANGE	FREQUENCY	CALIBRATION METHOD
Cont. Air Monitor	5	Variable	0-50K cpm 0-10 cfm	Annual with Quarterly Check	Manometer, Standard Reference Materials
DOSIMETERS ION CHAMBER					
Direct	As Required	Variable	0-200mR	Semiannual Check	Standard Reference Materials
Direct	10	Variable	0-1 R	Semiannual Check	Standard Reference Materials
Area Radiation Monitors (ARM)	Specified in Table 12.3-14			Annual with Quarterly Check	Standard Reference Materials

within the Technical Staff and executes the appropriate station responsibilities and rights called for in the fuel contracts and test agreements.

The LNE reviews applicable core reload, transient, and accident analyses.

Succession of Authority

The Station Superintendent has overall responsibility for station operation. He designates, in writing, which of the Assistant Superintendents is responsible in his absence. These assistants satisfy the requirements of ANSI 18.1-1971 for Plant Manager.

13.1.2.3 Operating Shift Crews

Position titles and applicable operator licensing requirements are identified in Table 13.1-2.

The minimum number of personnel planned for each shift are provided in Table 13.1-1.

Commencing no later than the loading of fuel in the first unit, a minimum of one Radiation/Chemistry Technician per shift will be available to provide around-the-clock coverage for implementation of the Radiation Protection Program. Additional personnel will be scheduled as required to cover special jobs or work loads as determined by the Radiation Protection Supervisor. During normal work days, the Radiation/Chemistry Technicians report to the Radiation Protection Supervisor. During off-shifts and weekends, the Radiation/Chemistry Technicians report to the Shift Engineer in the absence of the Radiation Protection Supervisor from the site.

13.1.3 Qualifications of Nuclear Plant Personnel

13.1.3.1 Qualification Requirements

The Byron/Braidwood Management follows the guidelines of ANSI N18.1 for personnel selection and training, except as specified in the following:

Rad/Chem Technicians at B/B will be qualified for this position as required by ANSI N18.1, 1971, except that individuals in training will perform work for which qualification has been demonstrated in order to obtain the experience required by the ANSI standard.

The B/B technical specifications require that the shift crew include an individual qualified in radiation protection procedures. An individual shall be considered qualified in radiation protection procedures upon certification by the licensee that he is capable of successfully accomplishing the following activities:

a. Conduct special and routine radiation, contamination and airborne radioactivity surveys and evaluate the results.

b. Establish protective barriers and post appropriate radiological signs.

c. Establish means of limiting exposure rates and accumulated radiation doses, including the use of protective clothing and respiratory protection equipment.

d. Perform operability checks of radiation monitors and survey meters.

e. Recommend appropriate immediate actions in the event of a radiological problem and perform necessary activities until the arrival of health physics personnel.

f. Conduct other routine radiological duties (e.g., Technical Specification surveillance items) as may be required on backshifts or weekends.

Individuals assigned to a shift crew who do not meet the experience requirements of ANSI N18.1, 1971, may perform radiation chemistry activities for which qualification has been demonstrated provided that the results of analyses performed by the individual are reviewed by:

a. a technician who meets the requirements of ANSI N18.1, 1971, Section 4.5.2, or

b. a supervisor not requiring an NRC license who meets the requirements of ANSI N18.1, 1971, Section 4.5.2, or

c. an individual who meets the requirements of ANSI N18.1, 1971, Section 4.4.3, "Radiochemistry."

Table 13.1-2 lists plant staff positions and designates ANSI N18.1 equivalent titles.

13.1.3.2 Qualifications of Plant Personnel

The qualifications of the initial staff personnel holding key managerial and supervisory positions are provided in the resumes included in Attachment 13A.

13A.21 RAD/CHEM SUPERVISOR - BYRON

Name: James R. Van Laeke

Education: Purdue University - 1971-1975
B.S. Environmental Health

Experience:

6/5/78 to Present:

Radiation Chemistry Supervisor, Radiation Chemistry Department, Byron Nuclear Station - Commonwealth Edison Company

Responsibilities: Radiation Protection, Chemistry and Radio-chemistry Programs. Reviews Radiological and Chemical Reports for internal distribution, U.S. Nuclear Regulatory Commission Reports, State of Illinois Reports, U.S. E.P.A. Reports, and other information required by state and regulatory agencies.

12/1/77 to 6/5/78:

Lead Health Physicist, Radiation Chemistry Department, Zion Nuclear Station - Commonwealth Edison Company.

Responsibilities: Implementation of the Radiation Protection Program.

6/1/76 to 12/1/77:

Health Physicist, Radiation Chemistry Department, Zion Nuclear Station - Commonwealth Edison Company.

Responsibilities: Radiation Monitoring Program, Refueling outage coordinator for Radiation Protection program.

6/1/75 to 6/1/76:

Health Physicist, Production Systems, Analysis Department - Commonwealth Edison Company,

Responsibilities: Assist in S ndardization of procedures equipment and policeis in radiation protection for three operating nuclear stations.

8/1/74 to 5/20/75:

Radiological Technician, Bionuclearics Department - Purdue University.

Responsibilities: Radiation surveys, exposure, and waste disposal for research at the university.

Additional Training:

Introduction to Power Plant Operations:

PWR Simulator - Westinghouse

Decision Analysis - Kepner Tregoe

Supervising for Results - Commonwealth Edison Company

Personal Information:

Date of Birth - 11/11/52

13A.22 RAD/CHEM SUPERVISOR - BRAIDWOOD

Name: Paul A. Corwin

Birth Date: December 25, 1943

Education:

Carthage College, Kenosha, Wisconsin - 1977
B.A., Biology and Natural Science.

College of Lake County, Grayslake, Illinois - 1974
A.S., Chemistry.

Experience:

1978 to Present:

Radiation-Chemistry Supervisor, Braidwood Station
Commonwealth Edison Company, Chicago, Illinois.

Supervisor in charge of Chemistry and Health Physics at a PWR facility under construction. Duties included staffing, training, equipment procurement and development of a chemistry control program for plant startup. Health Physics assignements included filing for the station's By-product and Special Nuclear Materials Licenses, development of a station Alara program and work with the State of Illinois on emergency planning. Additional assignments included, task analysis, standardization of forms, evaluation of new equipment, NPDES reports and procedure writing.

1975 to 1978:

Assistant Chemist, Zion Station
Commonwealth Edison Company, Chicago Illinois.

In charge of the hot and cold labs, sample systems and all chemistry department reports. Developed a technician training program in chemistry and lab safety, plus taught the courses. During refuelings my duties included supervision (foreman) of shift technicians, dose approvals and operation of the multi-channel analyzers.

1971 to 1975:

Radiation-Chemistry Technician, Zion Station
Commonwealth Edison Company, Chicago, Illinois.

Performed all routine chemistry and Health Physics tasks.

1965 - 1977:

Military:

1965 - 1977:

U.S. Navy, enlisted in the Nuclear Power Program, passed all training courses and qualified on all mechanical watch stations including engineering laboratory technician. Tour of duty included three years of instructor duty, plus one refueling at the D1G prototype. Separated from the service with a rank of E-6, under honorable conditions.

Affiliations:

The Health Physics Society

Health Physics Society, Midwest Chapter

Member of National Chapter Membership Committee

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION, REVIEW, INVESTIGATION, AND AUDIT

A. The Station Superintendent shall have overall full-time responsibility for safe operation of the facility. During periods when the Station Superintendent is unavailable, he shall designate this responsibility to an established alternate who satisfies the ANSI N18.1 experience requirements for plant manager.

B. The corporate management which relates to the operation of this station is shown in Figure 6.1-1.

C. The normal functional organization for operation of the station shall be as shown in Figure 6.1-2. The shift manning for the station shall be as shown in Table 6.1-1.

A fire brigade of at least five members shall be maintained on site at all times. The fire brigade shall not include the minimum shift crew necessary for safe shutdown of the plant (three members) or any personnel required for other essential functions during a fire emergency.

D. Qualifications of the station management and operating staff shall meet minimum acceptable levels as described in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971, with the exception of the Radiation/Chemistry Supervisor who whall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The individual filling the position of Assistant/Supervisor/Administrative and Support Services shall meet the minimum acceptable level for "Technical Manager" as described in Section 4.2.4 of ANSI N18.1-1971.

E. Retraining and replacement training of Station personnel shall be in accordance with ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971.

A training program for the fire brigade shall be maintained under the direction of the station fire marshall and shall meet or exceed the requirements of Section 27 of the NFPA Code - 1976.

F. Retraining shall be conducted at intervals not exceeding 2 years.

G. The Review and Investigative Function and the Audit Function of activities affecting quality during facility operations shall be constituted and have the responsibilities and authorities outlined below: