NUREG-0876
Supplement No. 4

Safety Evaluation Report
related to the operation of Byron Station, Units 1 and 2

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

Commonwealth Edison Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

May 1984

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555

2. The NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission,
   Washington, DC 20555

3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the NRC/GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission issuances*.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

GPO Printed copy price: $3.75

NUREG-0876
Supplement No. 4

Safety Evaluation Report
related to the operation of
Byron Station,
Units 1 and 2

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

Commonwealth Edison Company

U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

May 1984

ABSTRACT

Supplement No. 4 to the Safety Evaluation Report related to Commonwealth Edison Company's application for licenses to operate the Byron Station, Units 1 and 2, located in Rockvale Township, Ogle County, Illinois, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report.

TABLE OF CONTENTS

<u>TABLE OF CONTENTS</u> (continued)

APPENDICES

1 INTRODUCTION AND GENERAL DESCRIPTION OF FACILITY

1.1 Introduction

The Nuclear Regulatory Commission's Safety Evaluation Report (SER) (NUREG-0876) in the matter of Commonwealth Edison Company's application to operate the Byron Station Units 1 and 2 was issued in February 1982. The first supplement (SSER) to that SER was issued in March, 1982 the second was issued in January 1983, and the third was issued in November 1983. In the supplements, the staff identified items that were not yet resolved with the applicant. These items were categorized as

(1) Outstanding items which needed resolution prior to the issuance of an operating license.

(2) Items for which the staff had completed its review and had determined positions for which there appeared to be no significant disagreement between the applicant and the staff. Further information was needed, however, to confirm these positions.

(3) Items for which the staff had taken position and would require implementation and/or documentation after the issuance of the operating license. These would be conditions to the operating license.

The purpose of this fourth supplement to the SER is to provide the staff evaluation of the open items that have been resolved and to address changes to its safety evaluation that resulted from the receipt of additional information from the applicant.

The original SER stated that the staff had requested the applicant to verify that the Byron station meets the pertinent regulating requirements in 10 CFR 20, 50 and 100. The applicant responded by letter dated April 8, 1982 which closes confirmatory issue 36.

Copies of this SER supplement are available for inspection at the NRC Public Document Room, 1717 H Street, NW, Washington, D.C., and at the Rockford Public Library, Rockford, Illinois. Single copies may be purchased from the sources indicated on the inside front cover.

The NRC Project Manager assigned to the Operating License application for Byron Station is Leonard N. Olshan. Mr. Olshan may be contacted by calling (301) 492-7070 or writing:

Leonard N. Olshan
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

1.7 Summary of Outstanding Items

The following item are being closed in this supplement: emergency preparedness plans and facilities (Section 13.3), control room human factors review (Section 18.0). One outstanding item, inadvertently omitted in the original SER, has been added. The current status of the outstanding items listed in the original SER follows:

(1) Additional information to confirm pipeline foundation design (Section 2.5) Still open.
(2) Turbine missile evaluation (Section 3.5.1.3) - Still open.
(3) High- and moderate-energy pipe break analysis outside containment (Section 3.6.1) - Closed in Supplement 2.
(4) Pump and valve operability assurance (Section 3.9.3.2) - Still open.
(5) Baseplate flexibility and anchor bolt loading (Section 3.9.3.4) - Closed in Supplement 3.
(6) Seismic and dynamic qualification of equipment (Section 3.10) - Still open.
(7) Environmental qualification of electrical equipment (Section 3.11) - Still open.
(8) Improved thermal design procedures (Section 4.4.1) - Still open.
(9) TMI action item II.F.2: Inadequate Core Cooling Instrumentation (Section 4.4.7) - Still open.
(10) Steam generator flow-induced vibrations (Section 5.4.2) - Still open.
(11) Reactor pressure vessel forces and moments analysis (Section 6.2.1.2) - Closed in Supplement 2.
(12) Equipment and floor drainage system for internal flood protection (Section 9.3.3) - Closed in Supplement 2.
(13) Fire protection program (Section 9.5.1) - Partially closed in Supplement 3.
(14) Residual moisture in diesel air start piping (Section 9.5.6) - Closed in Supplement 1.
(15) Volume reduction system (Sections 11.1 and 11.4.2) - Still open.
(16) Emergency preparedness plans and facilities (Section 13.3) - Closed in this supplement.
(17) Control room human factors review (Section 18.0) - Closed in this supplement.
(18) Conformance of ESF filter system to RG 1.52 (Section 6.5.1) - Errata, added in this supplement.

1.8 Confirmatory Issues

Confirmatory items 12, 33, 35 and 36 from the original SER are being closed in this supplement. In addition, an interim solution for confirmatory issue 16 has been found acceptable. Two confirmatory issues, inadvertently omitted in the original SER, have been added. The current status of the confirmatory issues follows:

(1) Confirmatory analysis to verify river screenhouse seismic response analysis (Section 2.5.4.3) - Still open.
(2) Category 1 manhole protection from tornado missiles (Section 3.5.3) - Closed in Supplement 1.
(3) Analysis of tangential shear on containment (Section 3.8.1) - Errata, deleted in Supplement 2.

(4) Piping vibration test program (Section 3.9.2.1) - Still open.
(5) Snubber inspection and testing program details (Section 3.9.2.1) - Closed in Supplement 1.
(6) Seismic reevaluation of components and supports (Section 3.9.2.2) - Closed in Supplement 1.
(7) Basis for steam generator tube plugging (Section 3.9.3.1) - Closed in Supplement 3.
(8) Inservice testing of pumps and valves (Section 3.9.6) - Still open.
(9) Loose parts monitoring system (Section 4.4.6) - Closed in Supplement 2.
(10) Code cases for control valves (Section 5.2.1) - Closed in Supplement 1.
(11) Fracture toughness data for Bryon Unit 2 (Section 5.3.1) - Closed in Supplement 2.
(12) Steam generator tube surveillance (Section 5.4.2.2) - Closed in this supplement.
(13) Boration to cold shutdown analysis (Section 5.4.3) - Closed in Supplement 2.
(14) Cooldown rate with RHR (Section 5.4.3.1) - Closed in Supplement 2.
(15) RCS vent procedures (Section 5.4.5) - Closed in Supplement 2.
(16) Charging pump deadheading (Section 6.3.2), (Section 7.3.2.13) - Still open.
(17) Containment differential pressure analysis (Section 6.2.1) - Closed in Supplement 2.
(18) Containment sump instrumentation (Section 6.2.2.1) - Still open.
(19) Minimum containment pressure analysis for performance capabilities of ECCS (Section 6.2.1.5) - Still open.
(20) Containment leakage testing vent and drain provisions (Section 6.2.6) - Still open.
(21) Confirmatory test for sump design (Section 6.3.4.1) - Still open.
(22) Upper head temperature verification (Section 6.3.5.1) - Closed in Supplement 2.
(23) IE Bulletin 80-06 (Section 7.3.2.3) - Still open.
(24) Test jacks for P-4 interlock test (Section 7.3.2.9) - Closed in Supplement 2.
(25) Remote shutdown capability (Section 7.4.2.2) - Still open.
(26) Steam generator pressure control (Section 7.4.2.3) - Closed in Supplement 2.
(27) Switchover from injection to recirculation (Section 7.6.2.3) - Closed in Supplement 3.
(28) TMI Item II.K.3.1 (Section 7.6.2.7); III.D.1.1 (Section 9.3.5); II.K.2.17 (Section 15.5); II.D.I (3.9.3.3); II.K.2.17 - Closed in Supplement 2, others still open.
(29) Viewing the installation and arrangement of electrical equipment (Section 8.1) - Closed in Supplement 3.
(30) Independence of redundant electrical safety equipment (Section 8.4.4) -Closed in Supplement 2.
(31) Electrical distribution system voltage verification (Section 8.2.4) -Still open.
(32) Combined health physics and chemistry organization (Section 12.5.1) - Closed in Supplement 3.
(33) Revision to Physical Security Plan (Section 13.6) - Closed in this supplement.
(34) RCP rotor seizure and shaft break (Section 15.3.6) - Still open.
(35) Anticipated Transients Without Scram (ATWS) (Section 15.6) - Closed in this supplement.

(36) Applicant compliance with the Commission's regulations (Section 1.1) - Closed in this supplement.
(37) SWS process control program (Section 11.4.2) - Errata, added in this supplement.
(38) Noble gas monitor (Section 11.5.2) - Errata, added in this supplement.

1.9 License Conditions

License Conditions 4, 9, and 10 are being closed in this supplement. Four license conditions have been added.

Following is the current status of the license conditions:

(1) Groundwater monitoring program (Section 2.4.6)
(2) Masonry walls (Section 3.8.3)
(3) Preservice and Inservice inspection program (Sections 5.2.4 and 6.6)
(4) Response time testing (Section 7.2.2.5) - Closed in this supplement.
(5) Post accident monitoring (Section 7.5.2.2) - Closed in Supplement 2
(6) Modifications to permit isolation of non-IE loads from Class 1E power sources (Section 8.3.2) - Errata, deleted in Supplement 1.
(7) Compliance with Appendix R of 10 CFR 50, Fire Protection (Section 9.5.1)
(8) Steam valve inservice inspection (Section 10.2)
(9) Implementation of secondary water chemistry monitoring and control program as proposed by the Byron/Braidwood FSAR (Section 10.3.2) -Closed in this supplement.
(10) Personnel on shift with previous commercial PWR experience during startup phase (Section 13.1.2.1) - Closed in this supplement.
(11) TMI Item II.B.3 Postaccident Sampling (Section 9.3.2)
(12) Natural circulation testing (Section 5.4.3) - Errata, added in Supplement 3.
(13) Control of heavy loads (Section 9.1.5)
(14) Upgrade emergency operating procedures (Section 13.5.2)
(15) Relocate control room controls (Section 18.2)
(16) Emergency planning (Section 13.3)

5 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.4 RCPB Inservice Inspection and Testing

5.2.4.3 Evaluation of Compliance of Byron Unit 1 with 10 CFR 50.55a(g)

Several flaw indications were reported during the preservice examination of the vessel shell welds in the pressurizer and steam generators of Byron Station Unit 1 and determined to be rejectable by ASME Code Section XI criteria. Supplementary non-destructive examinations, e.g., ultrasonic examination, magnetic particle examination, and visual examination were performed from the inside diameter surface on the pressurizer and loop 2 steam generator to characterize the type and dimensions of the flaw indications. Radiographic examinations of the pressurizer and loop 2 steam generator were also performed and compared with the ultrasonic indications. Field examinations were conducted in accordance with the 1977 Edition of the ASME Code through the Summer 1977 Addenda.

Section XI of the ASME Code requires that components whose preservice examination reveals flaw indications that exceed the code allowable standards shall be unacceptable for service unless such flaws are removed or repaired to the extent necessary to meet the allowable flaw indication standards prior to placement of the components in service. Therefore, the Code does not provide procedures for the acceptance of preservice flaws that exceed the acceptance standards described in Article IWB-3000 based on fracture mechanics evaluation. However, methods and criteria are defined in Appendix A of Section XI for the fracture mechanics evaluation of flaw indications detected during inservice inspections.

In a letter dated October 27, 1983, the applicant described the results of the preservice non-destructive examinations of the steam generators and pressurizer. The applicant will remove the flaw indications in the steam generators and perform the repairs required by Section XI of the ASME Code. However, the applicant has determined that repair of the pressurizer shell weld is not warranted and requested that the NRC staff perform a plant specific review of the flaw indications. The applicant has characterized the flaw indication as fine-lines of embedded slag in the pressurizer lower head to lower circumferential shell weld, approximately T/4 (T, the shell thickness, is 4 inches) from the inside surface. The largest through thickness dimension is approximately 0.60 inches.

An ultrasonic examination of the region between 106" to 140" clockwise (CW) position was performed from the inside of the pressurizer based on the preservice examination results. The applicant has determined that all flaw indications between 106" to 135" CW position were below the ultrasonic response amplitude recording level described in Section XI and, therefore, these indications were dispositioned as acceptable. Supplemental ultrasonic examinations determined that a flaw indication above the ultrasonic recording level, approximately 2 1/8" in length, exists at the 140" CW position. The radiographic and ultrasonic testing from the I.D. of the pressurizer indicate that the ultrasonic reflectors apparently were amplified due to the geometry of the O.D. examination surface.

In the letter of October 27, 1983, the applicant reached the conclusion that repair activity could result in a metallurgical condition that is worse than the potential effects of retaining the flaw. The applicant requested that the NRC staff evaluate the results of fracture mechanics analysis assuming that the flaw indication is 3.1875" in length and 0.59" in the through wall direction. The fracture mechanics evaluation included all the possible thermal cycles and pressure loads that are anticipated during the lifetime of the plant. The calculations show that the flaw indication meets the acceptance criteria in Section XI, IWB-3600 if detected during an inservice inspection and, therefore, would not be required to be repaired. The analysis indicates that the flaw could grow approximately 35 mils during the life of the plant based on the crack growth curves included in Section XI.

Section XI of the ASME Code requires the removal of all flaw indications detected during preservice examinations that exceed the flaw acceptance standards defined in Article IWB-3000. The applicant intends to remove all unacceptable flaw indications in the steam generator shell welds and make the appropriate repairs required by Section XI of the ASME Code. The applicant requested that the NRC staff make a plant specific review of the flaw indications in the pressurizer weld that were characterized as embedded slag located approximately at T/4.

The staff review has determined that the applicant's fracture mechanics calculations were performed in accordance with Section XI procedures for an indication detected during inservice inspections. The staff has also determined that removal of the slag lines in the pressurizer and the associated field repair activities would be more detrimental to the integrity of pressurizer than retaining the flaws. Based on the above, the staff has determined the fracture mechanics evaluation presented in the applicant's letter dated October 27, 1983 is an acceptable justification for not performing the repairs required by Section XI of the ASME Code. However, during the review of the applicant's initial inservice inspection program after licensing, the staff will require that ultrasonic examinations be performed within the first inspection period in accordance with the requirements of the Code on the subject pressurizer shell weld to assure that the flaws have not grown beyond the calculated dimensions.

5.3 Reactor Vessel

5.3.2 Pressure-Temperature Limits

Pressure temperature limits must meet the requirements of Appendix G, 10 CFR 50. Revision to 10 CFR 50, Appendix G, was published in the Federal Register on May 27, 1983 and became effective on July 26, 1983. The amended Appendix G, 10 CFR 50 states that when pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions that are highly stressed by bolt preload must exceed the reference temperature of the materials in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests, unless a lower temperature can be justified by showing that the margins of safety for those regions, when they are controlling, are equivalent to those required for the beltline, when it is controlling.

By letter dated January 3, 1984 the applicant stated that the heatup and cooldown curves for Byron were reviewed and the cooldown curve for Byron 1 had to be revised to incorporate amended Appendix G limits. The staff has reviewed the revised curve submitted with the January 3, 1984 letter, and has found it acceptable.

5.4 Component and Subsystem Design

5.4.2 Steam Generators

5.4.2.2 Steam Generator Tube Inservice Inspection

The staff has completed its review of applicant's commitments to conform to the Westinghouse Standard Technical Specifications (NUREG-0452) in the area of steam generator tube inservice inspection, and finds them acceptable. Thus, Confirmatory Issue 12 is considered closed.

5.4.6 Seismic and Environmental Qualification of Pressurizer Power-Operated Relief Valves (PORVs)

In a letter dated October 7, 1982, the staff required the applicant to justify that the seismic and environmental design of the pressurizer power-operated relief valves (PORVs) will be appropriate for safety-related functions. Specifically, the staff questioned the use of the PORVs as high point vents in achieving cold shutdown and in mitigating steam generator tube ruptures. The applicant provided additional information concerning each of these functions in Amendment 43, response to Question 212.160, dated September 1983. The applicant stated that the PORV at Byron would be upgraded to operate following a safe shutdown earthquake (SSE). In addition, the applicant justified that the current environmental design of the PORV (service in a mild environment) is adequate.

The staff has concluded that the above issues are resolved.

High Point Vents

High point vents are required for noncondensible gas removal by 10 CFR 50.44 (c)(3)(iii). Specifically, venting capability is required for the reactor coolant system, the reactor vessel head, and other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause the loss of function of these systems. High point vents are not required, however, for the tubes in U-tube steam generators. Because severe accidents involving noncondensible gas generation might exceed the environmental design of the PORVs, the applicant does not rely on the PORVs as high point vents but instead relies on the reactor vessel head vents, which are provided with redundant valving in parallel paths. The staff's acceptance of the head vent design is discussed in Section 5.4.5 of the SER. Because the head vent design is adequate to remove noncondensible gas from the reactor system and because noncondensible gas accumulation in the pressurizer will not affect core cooling, the staff has concluded that the high point vent design for Byron is adequate without reliance on the PORVs.

Cold Shutdown Capability

Branch Technical Position (BTP) RSB 5-1, attached to Standard Review Plan Section 5.4.7, requires that the reactor be designed to be taken from normal operating conditions to cold shutdown using only safety-related equipment. Because the PORVs may be used to depressurize the reactor system during the approach to cold shutdown, the staff required that they be designed to operate following an SSE or that an alternative safety-related method for reactor system depressurization be provided. The PORVs will be upgraded to operate following an SSE; therefore, this matter is resolved. Environmental consideration for cold shutdown is the same as for a steam generator tube rupture (SGTR) as discussed below.

Steam Generator Tube Rupture

Use of the PORVs may be required to depressurize the reactor coolant system during SGTR events when the pressurizer level is maintained by the emergency core cooling system (ECCS). The staff required the applicant to demonstrate that the PORVs are seismically and environmentally qualified to mitigate this event or to provide other appropriately qualified safety-related equipment. The applicant calculated that the containment environment produced by an SGTR would be mild with a maximum containment temperature of 160°F. The temperature increase would be produced by opening the PORV assuming failure of the nonsafety-related pressurizer relief tank. The PORV block valves, which are environmentally qualified for harsh environments, could be relied on to isolate a stuck-open PORV. The staff concurs with the applicant's assessment that the environmental qualification of the PORVs is adequate. The applicant has committed to upgrade the seismic qualification of the PORVs to SSE; therefore, this matter is resolved.

6 ENGINEERED SAFETY FEATURES

6.3 Emergency Core Cooling System

6.3.2 Evaluation of Single Failures

The SER documented a commitment by the applicant to install an automatic system to ensure adequate minimum centrifugal charging pump flow to prevent deadheading that could damage the pumps in the ECCS mode of operation. The staff accepted the commitment subject to confirmation of the applicant's design. The details of the modification were submitted in a letter dated September 16, 1983. The modification is currently scheduled to be completed before startup at Byron Unit 2. For Byron Unit 1 the applicant will complete a portion of the modification and will provide for manual action to accomplish the remainder of the system functions until the first refueling outage. At that time the automatic system will be completed for Byron Unit 1.

In the original design the centrifugal charging pump bypass lines which are designed to protect the pumps from overheating at high pressures and low flows are automatically isolated by an SI signal. The bypass would not normally be needed for pump protection during LOCA events since reactor system pressures would be relatively low. Isolating the bypass lines provides an additional 8 lb/sec for core cooling.

For certain event sequences, such as a stuck open pressurizer relief valve which is subsequently isolated, charging pump deadheading might result with the present design. To protect the charging pumps from possible damage, the applicant will remove the SI closure signal from the bypass line valves, relocate the valves from a series to parallel arrangement so that single valve failure will not result in loss of bypass flow and provide for automatic closure of the bypass lines on low reactor system pressure (1400 psig) and SI. The valves would be automatically opened if the reactor system pressure increases above 2000 psig. Closure of additional isolation valves in the bypass lines will occur prior to switching to the recirculation mode of ECCS operation. Protection from deadheading would not be required in the recirculation mode since either the course of the accident or operator action will provide for reactor system depressurization before the Refueling Water Storage Tank could be emptied.

At Byron Unit 1, only a portion of the above modification will be completed during the first cycle. The bypass flow control valves 8110 and 8111 will be relocated from the series arrangement to the parallel arrangement before startup. The solenoid valves 8114 and 8116, however, will not be installed until the first refueling outage. Movement of the 8110 and 8111 valves before startup will have the benefit of reducing radiation exposure to plant personnel, rather than performing the entire modification during the first refueling.

The applicant will depend on operator action to control bypass flow at Byron 1 for the first cycle. The 8110 and 8111 valves would be manually closed if reactor system pressure decreased to 1400 psig and opened if reactor system pressure increased to 2000 psig.

Because the parallel isolation valve arrangement raises the possibility of single failure, the staff questioned the consequences of inadvertent bypass flow during the recirculation period when the charging pumps may pass contaminated sump water. Charging system pressure would be higher during the recirculation phase than during the injection phase and may cause the relief valves on the bypass lines to open. The applicant verified that the relief valve discharge would be contained within seismic category 1 equipment, thereby preventing release of radioactive material to the environment. The applicant also committed to implement operating procedures to provide for the manual isolation of the bypass line should the 8110 and 8111 valves fail to close during the recirculation phase of a LOCA.

The staff concludes that the interim solution to charging pump deadheading at Byron 1 is acceptable. However, Confirmatory Issue 16 will remain open until the design details of the permanent modification have been reviewed and approved by the staff.

6.4 Control Room Habitability

In FSAR Amendment 39, the applicant retracted the commitment to provide chlorine monitors in the control room ventilation system. In Section 2.2.3 of FSAR Amendment 43, the applicant provided justification for removal of these detectors by stating that there are no significant quantities of potentially hazardous chemicals stored at the plant site. In Section 2.2.1 of the original SER, the staff concluded that there are no transportation routes in the vicinity of the plant which could cause a chlorine hazard to the plant. Therefore, the staff concludes that the removal of the chlorine detectors for the Byron Station is acceptable.

7 INSTRUMENTATION AND CONTROL

7.2 Reactor Trip System

7.2.2 Specific Findings

7.2.2.5 Response Time Testing

By letter dated September 28, 1983, the applicant indicated that the report titled "The Use of Process Noise Measurement to Determine Response Characteristics of Protection Sensors in U.S. Plants" as submitted by Westinghouse to the Director of Licensing, NRC, on August 15, 1983, provides justification for the use of this technique for response time testing. The staff has reviewed the Westinghouse report which describes the test method and provides the results of tests conducted at operating reactors from 1977 through 1982 using this technique. The staff concludes that the use of process noise measurements will provide an acceptable means to fulfill the requirements for response time testing as specified in the plant Technical Specifications. Therefore, License Condition 4 is considered closed.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.5 Overhead Heavy-Load-Handling System

The SER stated that the applicant had committed to implement the interim actions described in the NRC generic letter dated December 22, 1980 prior to receiving an operating license. Staff review of the applicant's response to the generic letter is continuing. However, a condition will be placed in the license requiring that within six months of the issuance of the license, the applicant shall have implemented commitments acceptable to the staff regarding Phase I of the generic letter. Further, prior to startup following the second refueling outage, the applicant shall have made commitments acceptable to the staff regarding Phase II of the generic letter.

10 STEAM AND POWER CONVERSION SYSTEM

10.3 Main Steam Supply System

10.3.3 Secondary Water Chemistry

In the original SER, the staff found the applicant's secondary water chemistry monitoring and control program acceptable, and stated that the license would be conditioned to require that this program be carried out. The staff has since decided to incorporate this program in Section 6 (Administrative Controls) of the Technical Specifications which will be issued with the license. Thus, License Condition 9 is no longer necessary.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

13.1.2 Operating Organization

13.1.2.1 Organization

The original SER stated that the applicant had not committed to provide experienced individuals on each shift for at least one year from initial criticality, to include attainment of 100-percent power. By letter dated February 3, 1982 the applicant provided an acceptable commitment; thus, License Condition 10 is no longer necessary.

13.3 Emergency Planning

Since the issuance of Supplement No. 2 to Safety Evaluation Report, the applicant has conducted a full-scale offsite emergency preparedness exercise involving the State of Illinois and local counties within the 10 mile Emergency Planning Zone for Byron. In addition, by letters dated August 17 and December 15, 1982, from Mr. T. R. Tramm (CECo) to Mr. H. Denton, NRC, the applicant provided advanced information regarding the improvement items specified on pages D-21 and D-22 of the SER which were not addressed in Supplement No. 2. Finally, on January 13, 1984, the Atomic Safety and Licensing Board issued its Initial Decision regarding the licensing of Byron Station. Included in that decision were three license conditions related to emergency preparedness.

The purpose of this section is to update Appendix D of the Byron SER and Section 13.3 of Supplement 2 to the Byron SER by providing the staff evaluation of additional information. It is supplementary to and not in lieu of the discussion in Appendix D of the SER and Section 13.3 of Supplement 2. The material that follows is an update to the information contained in the same lettered subsections of Section 13.3 of Supplement 2.

ONSITE EMERGENCY PREPAREDNESS

D. Emergency Classification System

By letter dated August 17, 1982, the applicant committed to incorporate the EALs mentioned in the SER into the Byron Annex of the Generating Station Emergency Plan (GSEP). In addition, the applicant committed to providing specific parameter values where appropriate in the Annex. Although the Annex has not yet been revised to incorporate these commitments, the staff has reviewed the station emergency plan implementing procedure, BZP-200 and its associated table (BZP-200A1), entitled "Byron Emergency Action Levels." Based on this review, the staff concludes that specific parameter values for appropriate Emergency Action Levels (EALs) have been provided in the procedural table, and the EALs specified in Supplement 2 to the SER have been included in the procedural table. Based on the applicant's commitment to include this information in the next revision of the Byron Annex, the staff concludes that the applicant's provisions in this regard are adequate and Improvement Items 4 and 5 listed on page D-21 of the SER have been met. Therefore, the proposed license condition described in Subsection D on page 13-2 of Supplement 2 is no longer applicable.

E. Notification Methods and Procedures

In Supplement 2, the staff stated that it would await the Federal Emergency Management Agency (FEMA) findings with regard to the adequacy of the design of the installed prompt notification system. By letter dated February 1, 1984, from Mr. Richard W. Krimm, Assistant Associate Director of the Office of Natural and Technological Hazards Prgrams, FEMA, to Mr. Edward L. Jordan, Director of the Division of Emergency Preparedness and Engineering Response, NRC, FEMA stated that the administrative and physical means to notify the public within the ten mile EPZ through the use of outdoor sirens and indoor alert monitors have been provided for in planning. The hardware is in place. Assignment of responsibility for activating the system is the County Sheriff. Public alerting and notification and instructions for the public by use of the system were demonstrated during the November 15, 1983 full-scale exercise. The Emergency Broadcast System (EBS) messages included a descrption of affected areas. Sirens were sounded within ten minutes of the emergency action level broadcast.

FEMA has recently issued FEMA-43, "Standard Guide for the Evaluation of Alert and Notification Systems for Nuclear Power Plants," September 1983. An evaluation of the Byron system using this document has not been conducted. However, based on the review conducted to date, FEMA recommended an approval for this phase of licensing.

The staff concludes that the applicant's offsite provisions in regard to the prompt public alerting and notification system are adequate and Improvement Item 3 listed on page D-21 of the SER has been met.

G. Public Information

The applicant provided the staff with a copy of the proposed public information brochure by letter dated December 15, 1982. Comments were solicited from FEMA, and their comments along with those of the staff were provided to the applicant. In addition, the applicant included comments developed by the Intervenors in the final preparation of the brochure. The brochure was printed and distributed to members of the transient and permanent population during November 1983. Subsequently, the NRC's Region III Office conducted a detailed appraisal of the applicant's emergency preparedness program (see Inspection Reports No. 50-454/83-56 and 50-455/83-39). The results of this appraisal indicated that the brochure had been reviewed and distribution was coordinated wth appropriate State and local governmental agencies. The brochure had been distributed to the permanent and transient population within the ten mile plume exposure pathway Emergency Planning Zone. The staff concludes that the applicant's provisions in this regard are adequate and Improvement Item 7 listed on page D-21 if the SER has been met.

H. Emergency Facilities and Equipment

As discussed in Section H of Appendix D to the original SER (NUREG-0876) and Section H of Chapter 13.3 of Supplement 2 to the SER, the staff was continuing to review and evaluate a number of submittals from the applicant concerning its Emergency Response Facilities (ERFs). The staff has now completed its review and evaluations of the applicant's provisions for ERFs an determined that the ERFs are adequate as interim facilities. In December 1983, the NRC issued Supplement 1 to NUREG-0737 which contained final requirements and guidance for ERFs that superseded the previous requirements of items III.A.1.2 (Upgraded Emergency Support Facilities) and III.A.2.2 (Meteorological Date) of NUREG-0737. A determination of adequacy of the applicant's final ERFs will be performed during a post-implementation appraisal. That appraisal will be conducted against the provisions of Supplement 1 to NUREG-0737 at a future date. This is a confirmatory item and does not require resolution prior to licensing for full power operation. The staff considers that Improvement Item 8 on page D-21 of the SER is satisfactorily resolved.

J. Protective Response

By letter dated December 15, 1982, the applicant submitted their revised "Evacuation Time Estimates Within the Plum Exposure Pathway Emergency Planning Zone for the Byron Nuclear Generating Station." This revised evacuation time estimate was reviewed by the staff. In addition, contentions involving the evacuation time estimates were litigated before the Atomic Safety and Licensing Board (ASLB). The staff review and ASLB Initial Decision indicates that the evacuation time estimates do not currently address whether the preparation estimates include time to shut down employment sites. In addition, the thirty percent value used in estimating reduced capacity for adverse weather was viewed as being not appropriate for this locale by the ASLB. The following license conditions must be met prior to Unit 1 exceeding 5% of rated power:

(1) Applicant's Evacuation Time Study must be clarified, and amended if necessary, to reflect employment-center shutdown times.

(2) Applicant's Evacuation Time Study must be modified to reflect realistic time estimates under adverse weather conditions. Conservatisms may remain in the Study provided that they are clearly identified as such and quantified.

During the ASLB hearing, Illinois Department of Nuclear Safety officials stated that rather than use the average shielding value in determining whether evacuation or sheltering was the appropriate protective action, they would use the value representative of the least amount of shielding. This is inconsistent with guidance provided on page 64 of NUREG-0654, Revision 1. Consequently, the ASLB found that the use of a sheltering value representative of the least amount of shielding was inappropriate, and the following license condition is required to be met prior to Unit 1 exceeding 5% of rated power:

(1) The applicant shall provide information to the emergency planning officials, particularly the Illinois Department of Nuclear Safety, which realistically reflects the average sheltering values of the structures in the Byron plume exposure pathway EPZ.

N. Exercise and Drills

As discussed below, both the staff and members of the Federal Emergency Management Agency (FEMA) observed the November 15, 1933 full participation exercise for the Byron Plant. The staff has reviewed and evaluated the results of that exercise, including the February 1, 1984 FEMA Exercise Report, and determined that Improvement Item 10 listed on page D-21 of the SER has been met and is considered closed.

OFFSITE EMERGENCY PREPAREDNESS

The Federal Emergency Management Agency provided interim findings of the status of offiste preparedness in a letter dated February 1, 1984. The interim findings on the status of offsite preparedness are based on a review of the following: (1) Illinois Plan for Radiological Accidents (IPRA) State General Plan, Volume I, (2) IPRA State General Plan, Standard Operating Procedures (SOPs); (3) IPRA Byron Plan, Volume VI, Ogle County; (4) FEMA Region V Regional Assistance Committee (RAC) review of the IPRA Volumes 1 and VI and SOPs; (5) scenario and evaluation report for the full-scale exercise of November 15, 1983; and (6) the official notice, attendance roster, and transcript of the Public Meeting held on December 8, 1983, to discuss the planning for Byron.

As indicated in the Federal Emergency Management Agency's interim findings of February 1, 1984, and as demonstrated during he November 15, 1983 exercise, the Federal Emergency Management Agency states that the Illinois State and Ogle and Winnebago Counties offsite plans and preparedness are adequate to protect the health and safety of the public living in the vicinity of the Byron Nuclear Power Station and there is reasonable assurance that appropriate protective measures can be taken offsite in the event of a radiological emergency. Accordingly, the staff finds that the provisions of 10 CFR 50.47(A)(1) regarding offsite preparedness have been met. In the event that the NRC finds that the lack of progess in completion of the procedure in the FEMA rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of preparedness, the provisions of 10 CFR 50.54(s)(2) will apply. The license will be conditioned accordingly.

CONCLUSION

Subject to compliance with the above-identified license conditions, the staff concludes that Commonwealth Edison Company's Generating Station's Emergency Plan and Byron site-specific annex meet the planning standards of 10 CFR 50.47(b) and the requirements of 10 CFR 50, Appendix E, and conform with the guidance in NUREG-0654, Revision 1. Based on the above and review of the Federal Emergency Management Agency interim finding dated February 1, 1984, the staff concludes that the state of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

13.5 Plant Procedures

13.5.2 Operating and Maintenance Procedures

The applicant's programs for operating and maintenance procedures were previously reviewed and found acceptable with one item unreviewed: TMI Action Plan Item I.C.1, the re-analysis of accidents and transients and development of emergency operating procedures. This item was not reviewed because the applicant's program for emergency operating procedures had not been developed pending completion of generic technical guidelines by the Westinghouse Owners

Group (WOG) and their review by the NRC staff. The WOG Emergency Response Guideline Program was described to the staff in the WOG letters of November 30, 1981, July 21, 1982 and June 4, 1983. The staff's Safety Evaluation Report approving these guidelines for implementation was issued as Generic Letter 83-22, dated June 3, 1983.

In an earlier Generic Letter, 82-33, "Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability," dated December 17, 1982, the staff updated the requirements for programs to upgrade emergency operating procedures (EOPs). This implemented the staff's long-term plan for EOPs encompassing the requirements of TMI Action Items I.C.1, I.C.8, and I.C.9. This Generic Letter required licensees and applicants to submit, for staff review and approval, a Procedures Generation Package (PGP) to include (1) plant-specific technical guidelines, (2) a plant-specific writer's guide, (3) a description of the program to validate and verify the EOPs, and (4) a description of the training program for the EOPs.

In a letter dated April 14, 1983, the applicant responded to Generic Letter 82-33 for the Byron Station units. The letter stated that the PGP would be submitted 12 months after NRC approval of the first revision to the Westinghouse generic technical guidelines. Since this schedule would not support the licensing schedule, the applicant has developed EOPs based on the approved technical guidelines. The applicant still plans to update the EOPs and submit the PGP based on Revision 1 of the owners' group guidelines after the NRC reviews and approves the generic work. The staff finds this plan acceptable. Revision of the EOPs and submittal of the PGP will be made a condition of the operating license. The staff has reviewed the development of the EOPs that will be used until the EOPs based on Revision 1 are implemented. The results of this review are reported here. The staff will review the PGP under its program for review of operating reactor licensee responses to Generic Letter 82-33.

In letters from dated December 14, 1983 and January 5, 1984, the applicant provided information regarding the emergency operating procedures to be used during initial operation of Byron 1. This included the writer's guide (BAP 1310-A3 Revision 0), the instructions on the use of procedures (BAP 300-11 Revision 2), and the EOP validation and verification guide (File Number 1.2.103, HP, Basic, Revision 0). These materials were reviewed with the understanding that the procedures developed for initial operation will be used only for the time period necessary to develop and implement the full requirements of Generic Letter 82-33. The staff review of this material included an evaluation of the technical and human factors bases of the procedures as well as the operator training on the procedures.

The applicant's December 14, 1983 letter stated that the procedures developed were based upon the Westinghouse Owners Group generic guidelines approved by the staff. It also stated that these procedures were (1) reviewed by the Byron On-Site Review Board, (2) reviewed by representatives of Westinghouse (the NSSS vendor), and (3) evaluated through a formal validation and verification process. This process included table-top reviews, simulator exercises, and control room walk-throughs. Based on the description of the development program, the staff concluded that the interim EOPs have an acceptable technical basis.

The staff also reviewed how human factors principles were factored into the development of the interim EOPs. To ensure that the procedures are usable, accurate, convenient, readable, and acceptable to the control room personnel, the applicant developed a writer's guide, BAP 1310-A3. In accordance with the writer's guide, emergency operating procedures at Byron are in a two-column and two-level format. The left-hand column includes the operator actions and the expected plant responses in a two-level presentation. This includes, as the 'high level' step, what the operator must do and, as the 'low level' step, how to accomplish it. The right-hand column contains guidance for operator actions when the expected plant response is not obtained by the left-hand column steps. Other aspects of procedure writing as contained in NRC and industry guidance on the subject are also addressed by the writer's guide. The validation and verification process previously discussed was also intended to check that the EOPs conform to the writer's guide and that the EOPs are consistent with the control room design, minimum shift staffing, and the training program. The staff has concluded that human factors principles have been appropriately incorporated into the interim EOPs.

The letters of December 14, 1983 and January 5, 1984 also described training to be conducted to assure operator familiarity with the interim EOPs. This included training on the Emergency Operating Procedures including the Event Specific Subprocedures, the Emergency Contingency Actions, the Critical Safety Functions Status Trees, and the Function Restoration Procedures. The staff finds this training acceptable for use of the interim EOPs.

Based on (1) the review described above, (2) the commitments made by the applicant in letters dated April 14, 1983, December 14, 1983, and January 5, 1984, and (3) a license condition requiring implementation of EOPs based upon a Procedures Generation program as described in Generic Letter 82-33, the staff concludes that the applicant's plan for development and use of the emergency operating procedures is acceptable for licensing and operations up to and including 100 percent of rated power.

13.6 Industrial Security

Introduction

The applicant originally submitted for Byron the following security program plans which have since been revised and amended:

> "Byron Nuclear Power Station Physical Security Plan, Security Personnel Training and Qualification Plan, and Safeguards Contingency Plan", Revision 2 (May, 1980) transmitted by letter of May 2, 1980.

This Safety Evaluation Report (SER) summarizes how the applicant has provided for meeting the requirements of 10 CFR Part 73. The SER is composed of a basic analysis that is available for public review, and a protected Appendix. Based on a review of the subject documents and visits to the site, the staff has concluded that the protection provided by the applicant against radiological sabotage at the Byron Nuclear Station meets the requirements of 10 CFR Part 73. Accordingly, the protection will ensure that the health and safety of the public will not be endangered.

Physical Security Organization

To satisfy the requirements of 10 CFR 73.55(b), the applicant has provided a physical security organization that includes a Security Shift Supervisor who is onsite at all times with the authority to direct the physical protection activities. To implement the commitments made in the physical security plan, training and qualification plan, and the safeguards contingency plan, written security procedures specifying the duties of the security organization members have been developed and are available for inspection. The training program and critical security tasks and duties for the security organization personnel are defined in the "Byron Security Personnel Training and Qualification Plan" which meets the requirements of 10 CFR Part 73, Appendix B, for the training, equipping, and requalification of the security organization members. The physical security plan and the training program provide commitments that preclude the assignment of any individual to a security-related duty or task prior to the individual being trained, equipped, and qualified to perform the assigned duty in accordance with the approved guard training and qualification plan.

Physical Barriers

In meeting the requirements of 10 CFR 73.55(c) the applicant has provided a protected area barrier which meets the definition in 10 CFR 73.2(f)(1). An isolation zone, to permit observation of activities along the barrier, is provided as follows (except for the locations listed in the Appendix): at least 10 feet inside the inner perimeter fence, 20 feet between the two perimeter fences, and at least 5 feet outside of the outer fences.

The staff has reviewed those locations and determined that the security measures in place are satisfactory and continue to meet the requirements of 10 CFR 73.55(c).

Illumination of 0.2 foot-candles is maintained for the isolation zones, protected area barriers, and external portions of the protected area.

Identification of Vital Areas

The Appendix contains a discussion of the applicant's vital area program and identifies those areas and items of equipment determined to be vital for protection purposes. Vital equipment is located within vital areas which are located within the protected area and which require passage through at least two barriers, as defined in 10 CFR 73.2(f)(1) and (2), to gain access to the vital equipment. Vital area barriers are separated from the protected area barrier.

The control room and central alarm station are provided with bullet-resistant walls, doors, ceiling, floors, and windows. Based on these findings and the analysis set forth in paragraph C of the Appendix, the staff has concluded that the applicant's program for identification and protection of vital equipment satisfies the regulatory intent. However, this program is subject to onsite validation by the staff in the future, and to subsequent changes if found to be necessary.

Access Requirements

In accordance with 10 CFR 73.55(d) all points of personnel and vehicle access to the protected area are controlled. The individual responsible for controlling the final point of access into the protected area is located in a bullet-resistant structure. As part of the access control program, vehicles (except under emergency conditions), personnel, packages, and materials entering the protected area are searched for explosives, firearms, and incendiary devices by electronic search equipment and/or physical search.

Vehicles admitted to the protected area, except licensee-designated vehicles are controlled by escorts. Licensee-designated vechicles are limited to onsite station functions and remain in the protected area except for operational maintenance, repair, security, and emergency purposes. Positive control over the vehicles is maintained by personnel authorized to use the vehicles or by the escort personnel. A picture badge/key card system, utilizing encoded information, identifies individuals that are authorized unescorted access to protected and vital areas, and is used to control access to these areas. Individuals not authorized unescorted access are issued non-picture badges that indicate an escort is required. Access authorizations are limited to those individuals who have a need for access to perform their duties.

Unoccupied vital areas are locked and alarmed. During periods of refueling or major maintenance, access to the reactor containment(s) is positively controlled by a member of the security organization to assure that only authorized individuals and materials are permitted to enter. In addition, all doors and personnel/equipment hatches into the reactor containment(s) are locked and alarmed. Keys, locks, combinations, and related equipment are changed on an annual basis. I[illegible], when an individual's access authorization has been terminated du[illegible]k of reliability or trustworthiness, or for poor work performance [illegible] locks, combinations, and related equipment to which that person [illegible] are changed.

Detectio[illegible]

In satisfying the requirements of 10 CFR 73.55(e) the applicant has installed intrusion detection systems at the protected area barrier, at entrances to vital areas, and at all emergency exits. Alarms from the intrusion detection system annunciate within the continuously manned central alarm station and a secondary alarm station located within the protected area. The central alarm station is located such that the interior of the station is not visible from outside the perimeter of the protected area. In addition, the central station is constructed so that walls, floors, ceilings, doors, and windows are bullet-resistant. The alarm stations are located and designed in such a manner so a single act cannot interdict the capability of calling for assistance or responding to alarms. The central alarm station contains no other functions or duties that would interfere with its alarm response function. The intrusion detection system transmission lines and associated alarm annunciation hardware are self-checking and tamper-indicating. Alarm annunciators indicate the type of alarm and its location when activated. An automatic indication of when the alarm system is on standby power is provided in the central alarm station.

Communications

As required in 10 CFR 73.55(f) the applicant has provided for the capability of continuous communications between the central and secondary alarm station operators, guards, watchmen, and armed response personnel through the use of a conventional telephone system, and a security radio system. In addition, direct communication with the local law enforcement authorities is maintained through the use of a conventional telephone system and two-way FM radio links. All non-portable communication links, except the conventional telephone system, are provided with an uninterruptable emergency power source.

Test and Maintenance Requirements

In meeting the requirements of 10 CFR 73.55(g) the applicant has established a program for the testing and maintenance of all intrusion alarms, emergency alarms, communication equipment, physical barriers, and other security-related devices and equipment. Equipment or devices that do not meet the design performance criteria or have failed to otherwise operate will be compensated for by appropriate compensatory measures as defined in the "Byron Nuclear Power Station Physical Security Plan" and in site procedures. The compensatory measures defined in these plans will assure that the effectiveness of the security system is not reduced by failures or other contingencies affecting the operation of the security-related equipment or structures. Intrusion detection systems are tested for proper performance at the beginning and end of any period that they are used for security. Such testing will be conducted at least once every 7 days.

Communication systems for onsite communications are tested at the beginning of each security shift. Offsite communications are tested at least once each day.

Audits of the security program are conducted once every 12 months by personnel independent of site security management and supervision. The audits, focusing on the effectiveness of the physical protection provided by the onsite security organization implementing the approved security program plans, include, but are not limited to: a review of the security procedures and practices, system testing and maintenance programs, and local law enforcement assistance agreements. A report is prepared documenting audit findings and recommendations and is submitted to the plant management.

Response Requirements

In meeting the requirements of 10 CFR 73.55(h) the applicant has provided for armed responders immediately available for response duties on all shifts consistent with the requirements of the regulations. Considerations used in support of this number are attached (see Appendix). In addition, liaison with local law enforcement authorities to provide additional response support in the event of security events has been established and documented.

The applicant's safeguards contingency plan for dealing with thefts, threats, and radiological sabotage events satisfies the requirements of 10 CFR Part 73, Appendix C. The plan identifies appropriate security events which could initiate a radiological sabotage event and identifies the applicant's preplanning, response resources, safeguards contingency participants, and coordination activities for each identified event. Through this plan, upon the detection of abnormal presence or activities within the protected or vital areas, response activities using the available resources would be initiated. The response activities and objectives include the neutralization of the existing threat by requiring the response force members to interpose themselves between the adversary and their objective, instructions to use force commensurate with that used by the adversary, and authority to request sufficient assistance from the local law enforcement authorities to maintain control over the situation.

To assist in the assessment/response activities a closed circuit television system, providing the capability to observe the entire protected area perimeter, isolation zones and a majority of the protected area, is provided to the security organization.

Employee Screening Program

In meeting the requirements of 10 CFR 73.55(a) to protect against the design basis threat as stated in 10 CFR 73.1(a)(1)(ii), the applicant has provided an employee screening program. Personnel who successfully complete the employee screening program or its equivalent may be granted unescorted access to protected and vital areas at the Byron site. All other personnel requiring access to the site are escorted by persons authorized and trained for escort duties and who have successfully completed the employee screening program. The employee screening program is based upon accepted industry standards and includes a background investigation, a psychological evaluation, and a continuing observation program. In addition, the applicant may recognize the screening program of other nuclear utilities or contractors based upon a comparability review conducted by the applicant.

The plan also provides for a "grandfather clause" exclusion which allows recognition of a certain period of trustworthy service with the utility or contractor, as being equivalent to the overall employee screening program. The staff has reviewed the applicant's screening program against the accepted industry standards (ANSI N18.17 1973) and has determined that the program is acceptable.

15 ACCIDENT ANALYSES

15.2 Normal Operations and Operational Transients

15.2.4 Changes in Core Reactivity Transients

15.2.4.3 Rod Cluster Control Assembly Malfunctions

The original SER indicated that a potential controller problem existed for the dropped control rod event which could lead to the imposition of operating restrictions. It also indicated that a detailed analysis would probably show that if the problem occurs that thermal limits would not be exceeded. Since then Westinghouse has developed a solution for the problem via a new methodology for analyzing the event and has documented it in a topical report (WCAP-10297P). This report and its methodology have been approved by the staff. The solution requires a reactor-cycle specific analysis showing that DNB limits will not be exceeded. FSAR Amendment 44 includes a discussion of this analysis, and the results for cycle one operation indicate that DNB limits will be met for this cycle. Thus operating limits will not be necessary for cycle one. Each future reload cycle will require similar cycle specific analysis as part of the normal reload analysis.

15.6 Anticipated Transients Without Scram

The Westinghouse generic guidelines approved by the staff in Generic Letter 83-22, dated June 3, 1983, adequately address the subject of Anticipated Transients Without Scram. As discussed in Section 13.5.2 of this supplement, the staff concludes that the applicant's implementation of these Westinghouse generic guidelines is acceptable for licensing and full power operation. Therefore, Confirmatory Issue 35 is considered closed.

18 HUMAN FACTORS ENGINEERING

18.2 Main Control Room and Remote Shutdown Panel

The original SER described the applicant's Preliminary Design Assessment (PDA) dated November 12, 1981, the staff's onsite Control Room Design Review/Audit (CRDR/A) of Novmeber 1981, and the CRDR/A report transmitted to the applicant January 11, 1982. The applicant developed resolutions to Human Engineering Discrepancies (HEDs) and proposed schedules for implementation of control room improvements. These were transmitted to the NRC by letter of May 9, 1983. Both reports were reviewed by the staff and several issues were clarified with the applicant during a meeting in Bethesda on October 26, 1983. A final submittal by the applicant documenting all of the clarifications was transmitted by letter dated January 6, 1984. All but one HED have been resolved.

The one unresolved item involves the relocation of the range and volume controls for the SOURCE RANGE nuclear instrument from the nuclear instrument cabinet 1PM07J to the main control board 1PM05J where they are needed during startup. The applicant has proposed to move the controls from panel 2PM07J to 2PM05J prior to its preoperational test. If test results indicate no technical problems, such as electrical noise interference, the change will be made on Unit 1 prior to completion of its first refueling outage. The staff agrees with this resolution and will make it a condition of the Unit 1 operating license.

All proposed control room improvements and the schedule for implementation are satisfactory to the staff. The staff concludes that, with these improvements, the potential for operator error leading to serious consequences as a result of human factors considerations in the control room will be sufficiently low to permit safe operation of Byron Station, Unit 1.

This completes the prelicensing staff evaluation of the Byron control room and the preliminary design assessment (PDA) portion of the TMI Action Plan, Item I.D.1. Thus, Outstanding Item 17 is closed. The plant must still be subjected to a detailed control room design review (DCRDR). Requirements for the DCRDR are identified in Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability," dated December 17, 1982. In addition, the DCRDR for Byron must address all PDA issues which the staff agreed could be postponed until that review. The DCRDR will be made a condition of the operating license and results will be reported in the future.

APPENDIX A

CHRONOLOGY OF NRC STAFF RADIOLOGICAL REVIEW OF BYRON STATION

The Safety Evaluation Report, Appendix A, provided a chronology of the NRC staff's radiological safety review for the period from April 30, 1977 to January 22, 1982. The Appendix A in Supplement No. 1 to the Safety Evaluation Report continued the chronology through March 22, 1982. Supplement No. 2 updated the chronology for Appendix A through November 19, 1982. Supplement No. 3 updated the chronology through September 16, 1983. This Supplement, No. 4, commences where Supplement No. 3 ended at September 16, 1983 and updates through March 30, 1984.

September 16, 1983	Letter from applicant concerning charging pump deadheading.
September 23, 1983	Letter from applicant concerning fuel load date.
September 23, 1983	Letter from applicant transmitting Amendment 43 to the FSAR.
September 28, 1983	Letter from applicant transmitting the Certificate of Service for Amendment 43 to the FSAR.
October 12, 1983	Letter to applicant concerning quality assurance and design verification program.
October 24, 1983	Letter to applicant concerning Rod Swap Analysis Method.
October 26, 1983	Representatives from NRC and CECO meet in Bethesda, MD to discuss control room human factors review (Summary issued November 7, 1983).
October 27, 1983	Letter from applicant concerning interim operation of HVAC Systems.
October 27, 1983	Letter from applicant concerning preservice inspection.
October 28, 1983	Representatives from NRC, CE, and S&L meet in Westinghouse offices in Bethesda, MD to discuss Byron fire protection program (Summary issued November 7, 1983).
October 28, 1983	Letter from applicant concerning Security Plan Revisions.
October 31, 1983	Letter to applicant concerning request for additional information - revised licensed operator requal topical report.
October 31, 1983	Letter to applicant concerning clarification of required actions based on generic implications of Salem ATWS events (Generic Letter 83-28).

November 1, 1983	Letter to applicant concerning staff evaluation of the modification to Westinghouse D4/D5/E Steam Generators.
November 3, 1983	Letter to applicant concerning the Emergency Operations Facilities.
November 5, 1983	Letter from applicant concerning a response to Generic Letter No. 83-28.
November 7, 1983	Letter from applicant concerning control of heavy loads.
November 17, 1983	Representatives from NRC and CECo meet in Bethesda, MD to discuss turbine maintenance program for the turbine missile issue (Summary issued December 6, 1983).
November 17, 1983	Letter from applicant concerning Security Plan Revisions.
November 18, 1983	Letter to applicant concerning Byron Station Physical Security Plan - Vital Equipment.
November 28, 1983	Letter to applicant transmitting Supplement No. 3 to the Byron Station SER (2 Xerox copies were sent).
December 5, 1983	Letter from applicant concerning masonry walls.
December 5, 1983	Letter from applicant concerning process control program.
December 6, 1983	Letter from applicant concerning dropped rod reanalysis.
December 6, 1983	Letter from applicant concerning preservice inspection program plan.
December 6, 1983	Letter from applicant concerning revised commitment regarding NUREG-0737 Supplement 1, Generic Letter 82-33.
December 7, 1983	Letter to applicant transmitting 20 copies of Supplement 3 to the Byron SER.
December 13, 1983	Letter to applicant requesting information needed prior to fuel load of Byron 1.
December 15, 1983	Letter to applicant concerning request for additional information - improved thermal design procedures.
December 15, 1983	Letter from applicant concerning revised commitment regarding NUREG-0737 Supplement 1, Generic Letter No. 82-33.
December 16, 1983	Letter from applicant concerning steam generator tube vibration.
December 19, 1983	Letter from applicant concerning charging pump deadheading.

December 20, 1983	Letter from applicant transmitting Amendment No. 44 to the FSAR.
December 22, 1983	Letter to applicant concerning interim operation of HVAC systems.
December 23, 1983	Letter from applicant concerning implementation of 10 CFR 61 and 10 CFR 20.311.
December 27, 1983	Letter from applicant concerning instrumentation for the detection of inadequate core cooling.
December 27, 1983	Letter from applicant concerning automatic PORV isolation.
December 28, 1983	Letter from applicant concerning 40-year operating license.
December 29, 1983	Letter from applicant concerning NUREG-0737 Supplement 1 SPDS Safety Analysis.
December 30, 1983	Letter from applicant concerning system leakage monitoring.
December 30, 1983	Letter from applicant concerning noble gas monitors.
December 30, 1983	Letter from applicant concerning Byron Station Security Plan.
December 30, 1983	Letter from applicant concerning pressurizer safety and relief valve.
January 3, 1984	Letter from applicant concerning diesel generator controls.
January 3, 1984	Letter from applicant concerning reactor vessel temperature limits.
January 4, 1984	Letter from applicant concerning supplemental response to Generic Letter 83-10c and d.
January 5, 1984	Letter from applicant concerning procedure generation package.
January 5, 1984	Letter from applicant concerning post accident sampling system.
January 5, 1984	Letter from applicant concerning postaccident sampling capability.
January 6, 1984	Letter from applicant concerning completion of preoperational test program.
January 6, 1984	Letter from applicant concerning Control Room PDA.
February 2, 1984	Letter to applicant concerning comments on the proposed offsite dose calculation model for Byron.

February 10,1984	Letter from applicant transmitting Supplemental information to the resolution of control of heavy loads at nuclear power plants.
February 17, 1984	Letter from applicant concerning use of penetrameter shims during radiography.
February 17, 1984	Letter from applicant concerning preservice inspection program plan.
February 21, 1984	Letter to applicant concerning deletion of home telephone numbers, unlisted utility numbers, etc. from emergency plans.
February 21, 1984	Letter from applicant transmitting the 1983 Annual Report.
February 22, 1984	Letter from applicant concerning Hydrogen Recombiners.
February 22, 1984	Letter from applicant concerning effects of local intense precipitation.
February 22, 1984	Letter from applicant concerning masonry walls.
February 22, 1984	Letter from applicant concerning GDC 51 Compliance Review.
February 22, 1984	Letter from applicant concerning system leakage monitoring.
February 28, 1984	Letter from applicant concerning preservice testing of snubbers.
February 28, 1984	Letter from applicant concerning river screenhouse seismic design.
March 21, 1984	Letter from applicant concerning response to Generic Letter 83-35 - Clarification of TMI item II.K.3.31 (SB LOCA Analyses).
March 22, 1984	Letter to applicant requesting additional information - fire protection and masonry walls.
March 30, 1984	Letter to applicant concerning Mechanical Equipment Environmental Qualification Program for Byron/Braidwood Stations.

APPENDIX F

NRC STAFF CONTRIBUTORS AND CONSULTANTS

The Supplement No. 4 to the SER is a product of the NRC staff. The following NRC staff members were principal contributors to this report.

NAME	TITLE	REVIEW BRANCH
M. Hum	Materials Engineer	Materials Engineering
S. Bhatt	Materials Engineer	Materials Engineering
D. Smith	Materials Engineer	Materials Engineering
B. Elliot	Materials Engineer	Materials Engineering
W. Jensen Sr.	Nuclear Engineer	Reactor Systems
F. Akstulewicz	Nuclear Engineer	Accident Evaluation
T. Dunning	Section Leader	Information & Control Systems
J. Wermeil	Section Leader	Auxiliary Systems
W. Kennedy	Sr. Operational Safety Engineer	Procedures and Systems Review
R. Skelton	Safeguards Analyst	Power Reactor SG Licensing
R. Eckenrode	Human Factors Engineer	Human Factors Engineering
H. Richings	Sr. Reactor Physicist	Core Performance
M. Phillips	Emergency Preparedness Inspection, Region III	Emergency Preparedness Licensing

APPENDIX G

ERRATA TO BYRON SAFETY EVALUATION REPORT

Page	Line	Change
1-14	5	Change "(Sections 10.2 and 10.4.2)" to "(Section 10.2)"
1-11	After last line	Add "(18) Conformance of ESF filter system to RG 1.52 (Section 6.5.1)"
1-13	After line 13	Add "(37) SWS process control program (Section 11.4.2), (38) Noble gas monitor (Section 11.5.2)"

NRC FORM 335
(6-83)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)	NUREG-0876 Suppl. #4
2. Leave blank	
3. TITLE AND SUBTITLE	Safety Evaluation Report related to the operation of Byron Station, Units 1 and 2
4. RECIPIENT'S ACCESSION NUMBER	
5. DATE REPORT COMPLETED	MONTH May, YEAR 1984
6. AUTHOR(S)	
7. DATE REPORT ISSUED	MONTH May, YEAR 1984
8. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)	Division of Licensing Office of Nuclear Reactor Regulation U. S. Nuclear Regulatory Commission Washington, D. C. 20555
9. PROJECT/TASK/WORK UNIT NUMBER	
10. FIN NUMBER	
11. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)	Division of Licensing Office of Nuclear Reactor Regulation U. S. Nuclear Regulatory Commission Washington, D. C. 20555
12a. TYPE OF REPORT	Safety Evaluation Report
12b. PERIOD COVERED (Inclusive dates)	November 1983- May 1984
13. SUPPLEMENTARY NOTES	Docket Nos. STN 50-454 and STN 50-455

14. ABSTRACT (200 words or less)

Supplement No. 4 to the Safety Evaluation Report related to Commonwealth Edison Company's application for licenses to operate the Byron Station, Units 1 and 2, located in Rockvale Township, Ogle County, Illinois, has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. This supplement reports the Status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report and Supplements 1 through 3.

15a. KEY WORDS AND DOCUMENT ANALYSIS

15b. DESCRIPTORS

16. AVAILABILITY STATEMENT	Unlimited
17. SECURITY CLASSIFICATION (This page)	UNCLASSIFIED
18. NUMBER OF PAGES	
19. SECURITY CLASSIFICATION (This report)	UNCLASSIFIED
20. PRICE	$

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

FIRST CLASS MAIL
POSTAGE & FEES PAID
USNRC
WASH D C
PERMIT No. G67

120555078877 1 1AN
US NRC
ADM-DIV OF TIDC
POLICY & PUB MGT BR-PDR NUREG
W-501
WASHINGTON DC 20555

NUREG-0876, Supp. No. 4 SER RELATED TO THE OPERATION OF BYRON STATION, UNITS 1 AND 2 MAY 1984