ISHAM, LINCOLN & BEALE
COUNSELORS AT LAW

THREE FIRST NATIONAL PLAZA
CHICAGO, ILLINOIS 60602
TELEPHONE 312 558-7500
TELEX: 2-5288

EDWARD S. ISHAM 1872-1902
ROBERT T. LINCOLN 1872-1889
WILLIAM G. BEALE 1885-1923

WASHINGTON OFFICE
1120 CONNECTICUT AVENUE, N.W.
SUITE 840
WASHINGTON, D.C. 20036
202 833-9730

DOCKETED

'82 JUN 24 AIO:30

June 21, 1982

Marshall E. Miller, Esq., Chairman
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. A. Dixon Callihan
Union Carbide Corporation
P.O. Box Y
Oak Ridge, Tennessee 37830

Dr. Richard F. Cole
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: In the Matter of Commonwealth Edison Company (Byron Stations, Units 1 and 2) Docket Nos. 50-454 & 50-455

Dear Administrative Judges:

Enclosed are copies of the following recent correspondence between the NRC Staff and Edison pertaining to Byron Station:

1) May 26, 1982, letter from W.S. Little to Cordell Reed regarding report 50-454/82-06 regarding inspection at Byron on May 3-6, 1982.

2) I.E. Bulletin No. 82-02 "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants"; R.C. DeYoung letter to All OLs and CPs dated June 2, 1982.

3) June 7, 1982, letter from T.R. Tramm to H.R. Denton regarding Byron/Braidwood loss of flow transients.

DS03

4) June 10, 1982, letter from B.J. Youngblood to L. DelGeorge regarding Byron/Braidwood reports on loose parts monitoring systems design and baseline testing.

5) June 14, 1982 letter from T.R. Tramm to H.R. Denton transmitting revision to Byron/Braidwood safe shutdown report.

6) I.E. Information notice no. 82-17: "overpressurization of reactor coolant system" Edward L. Jordan to All OLs and CPs dated June 11, 1982.

Sincerely,

Alan Bielawski
One of the attorneys for
Commonwealth Edison

cc: Service List

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

MAY 26 1982

I. R Report
50-454/82-06

file

Docket No. 50-454

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the routine safety inspection conducted by Mr. M. A. Ring of this office on May 3-6, 1982, of activities at Byron Nuclear Power Station, Unit 1, authorized by NRC Construction Permit No. CPPR-130 and to the discussion of our findings with Mr. R. Querio and others of your staff at the conclusion of the inspection.

The enclosed copy of our inspection report identifies areas examined during the inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

No items of noncompliance with NRC requirements were identified during the course of this inspection.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room. If this report contains any information that you (or your contractors) believe to be exempt from disclosure under 10 CFR 9.5(a)(4), it is necessary that you (a) notify this office by telephone within ten (10) days from the date of this letter of your intention to file a request for withholding; and (b) submit within twenty-five (25) days from the date of this letter a written application to this office to withhold such information. If your receipt of this letter has been delayed such that less than seven (7) days are available for your review, please notify this office promptly so that a new due date may be established. Consistent with Section 2.790(b)(1), any such application must be accompanied by an affidavit executed by the owner of the information which identifies the document or part sought to be withheld, and which contains a full statement of the reasons which are the bases for the claim that the information should be withheld from public disclosure. This

section further requires the statement to address with specificity the considerations listed in 10 CFR 2.790(b)(4). The information sought to be withheld shall be incorporated as far as possible into a separate part of the affidavit. If we do not hear from you in this regard within the specified periods noted above, a copy of this letter and the enclosed inspection report will be placed in the Public Document Room.

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

W. S. Little

W. S. Little, Chief
Engineering Inspection Branch

Enclosure: Inspection Report
No. 50-454/82-06(DETP)

cc w/encl:
Louis O. DelGeorge, Director
of Nuclear Licensing
V. I. Schlosser, Project Manager
Gunner Sorensen, Site Project
Superintendent
R. E. Querio, Station
Superintendent
DMB/Document Control Desk (RIDS)
Resident Inspector, RIII Byron
Resident Inspector, RIII
Braidwood
Mary Jo Murray, Office of
Assistant Attorney General
Myron M. Cherry

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No. 50-454/82-06(DETP)

Docket No. 50-454 License No. CPPR-130

Licensee: Commonwealth Edison Company
Post Office Box 767
Chicago, IL 60690

Facility Name: Byron Station, Unit 1

Inspection At: Byron Site, Byron, IL

Inspection Conducted: May 3-6, 1982

Inspector: M. A. Ring — L. N. Jackiw for — 5/20/82

Approved By: L. N. Jackiw, Chief, Test Program Section — L. N. Jackiw — 5/20/82

Inspection Summary

Inspection on May 3-6, 1982 (Report No. 50-454/82-06(DETP))
Areas Inspected: Routine unannounced inspection to review preoperational test procedures, witness the performance of preoperational testing, review the evaluation of preoperational test results, and review previous open items. The inspection involved 30 inspector-hours onsite by one NRC inspector including 0 inspector-hours onsite during off-shifts.
Results: No items of noncompliance were identified.

DETAILS

1. Persons Contacted

*R. Querio, Station Superintendent
*C. Tomashek, Startup Group
*D. St. Clair, Technical Staff Supervisor
*R. Ward, Assistant Superintendent, Administration and Support
*R. Pleniewicz, Assistant Superintendent, Operations
*A. Chomacke, Assistant Tech Staff Supervisor
*R. Westberg, QA Engineer, Operations
*R. Schwartz, QA Engineer, Construction

*Denotes those attending the exit interview.

Additional station technical and administrative personnel were contacted by the inspector during the course of the inspection.

2. Licensee Action on Previous Inspection Findings

a. (Open) Open item (454/82-02-01): This item involves providing acceptance values and tolerance ranges for data such as alarms and trip points being checked by a test (other than those values addressed as acceptance criteria in the FSAR). The inspector reviewed a Tech Staff Supervisor memo describing the licensee's method of providing these values. The item remains open pending review of the implementation of this method.

3. Preoperational Test Results Evaluation

The inspector reviewed the results of Preoperational Test 2.73.11 Safety Injection Accumulator System to verify the following attributes:

a. All test changes approved in accordance with administrative procedures, annotated in the procedures, and completed without changing the basic objectives of the test.

b. Deficiencies resolved, resolution accepted by appropriate management, retest requirements completed.

c. Engineering evaluation of results, agreement that testing demonstrated the system met design requirements, comparison with established acceptance criteria.

d. Data sheets completed, data within acceptance tolerances, deficiencies identified, test steps properly initialed and dated.

e. Quality assurance audit performed.

f. Results approved in accordance with administrative procedures.

The inspector reviewed the test following completion of the Test Review Board review, but prior to Project Engineering review, consequently, attribute c. above concerning Engineering evaluation is not complete and will be treated as an open item (454/82-06-01) pending completion of Engineering evaluation and subsequent inspector review. The inspector noted that the test procedure did not clearly show how the calculations of L/D in the procedure demonstrated the objective stated in Table 14.2.31 of the FSAR that "flow rate is as expected" or the acceptance criteria of Table 14.2.31 that "blowdown response is conservative with respect to the value used in the safety analysis." The licensee replied that the correlation between calculation results and FSAR criteria would be provided by the Project Engineering Department. This is an open item (454/82-06-02) pending Engineering evaluation and subsequent review by the inspector. This test also was approved by the Test Review Board prior to issuance of the Technical Staff Supervisor memo referenced in Paragraph 1 of this report and dealing with acceptance values and tolerance ranges for data other than those addressed as acceptance criteria in the FSAR. Consequently, the Safety Injection Accumulators did not have tolerance ranges specified for this "secondary" class of acceptance criteria. The licensee intends to reassess the procedure with respect to the "secondary" class of acceptance criteria and this is an open item (454/82-06-03) pending inspector review.

The inspector noted that Section 10.0 of this test provides for verification of operating procedures, however, most of the operating procedures were not available at the time of test performance and consequently not verified. The inspector questioned the licensee's method of ensuring that those operating procedures which require verification during the test program will receive verification and that those operating procedures not available at time of test procedure performance will be examined for possible verification by later test procedures. This is an open item (454/82-06-04) pending additional inspector review.

At the time of performance of 2.73.11, the licensee's Turnover/Testing Deficiency documentation form did not contain provisions for retest determination and signoff. Subsequently, the Startup Manual and the Turnover/Testing Deficiency form have been updated to require retest evaluation and to provide a retest signature. The evaluation of deficiencies associated with tests completed prior to the revision to the Startup Manual and the Turnover/Testing Deficiency form is considered an unresolved item (454/82-06-05) pending inspector review to ensure adequate retest.

No items of noncompliance were identified.

4. Preoperational Test Procedure Review

The inspector reviewed test Procedure 2.18.10 Chemical and Volume Control - VCT and Charging Pumps against the FSAR, SER and Regulatory Guide 1.68. The inspector made several comments against the procedure

which the licensee agreed to review for incorporation. This is an open item (454/82-06-06) pending further discussion with the licensee.

No items of noncompliance were identified.

5. Preoperational Test Witnessing and Related Items

During the course of witnessing portions of the Diesel Generator Preoperational Test 2.22.10, the inspector noted and followed up on the following related items.

a. The control room meter monitoring Diesel Generator output in kilowatts appeared to be inadequately sized since the meter was a 0-6000 kw full range meter while the Diesel Generator had a 6050 kw two hour load rating. The inspector's review found the inadequately sized meter to be documented in the licensee's deficiency system.

b. The inspector reviewed certificate of conformance records to determine that adequate documentary evidence was available at the site to show conformance to procurement requirements for diesel generator type qualification testing as described in IEEE-387.

c. The 1A Diesel Generator suffered a "crankcase explosion" in March of 1982. The inspector attempted to review the circumstances surrounding the event and the assessment of reportability with respect to 10 CFR 50.55(e). The licensee stated that Project Engineering had made the evaluation and that specifics of the assessment were not available at the site. This is an open item (454/82-06-07) pending review of the evaluation with Project Engineering.

6. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, Items of Noncompliance, or Deviations. An unresoloved item disclosed during the inspection is discussed in Paragraph 3.

7. Exit Interview

The inspector met with licensee representatives denoted in Paragraph 1 at the conclusion of the inspection on May 6, 1982. The inspector summarized the scope of the inspection and the findings. The licensee acknowledged the statements made by the inspector with respect to the open items and the unresolved item.

SSINS No.: 6820
IEB 82-02
OMB NO: 3150-0086
Expiration Date: 5/30/86

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

JUN 9 [illegible]

June 2, 1982

IE BULLETIN NO. 82-02: DEGRADATION OF THREADED FASTENERS IN THE REACTOR COOLANT PRESSURE BOUNDARY OF PWR PLANTS

Addressees:

All pressurized water nuclear power reactor facilities holding an operating license (OL), for action. All other nuclear power reactor facilities holding an operating license or construction permit (CP), for information.

Purpose:

The purpose of this bulletin is to: (1) notify licensees and construction permit holders about incidents of severe degradation of threaded fasteners (bolts and studs) in closures in the reactor coolant pressure boundary (RCPB), and (2) to require appropriate actions. A response to this bulletin is required from pressurized water reactors (PWRs) holding an operating license as discussed below.

Description of Circumstances:

In May 1980, Omaha Public Power District (OPPD) submitted a special maintenance report to the NRC about the significant corrosion wastage experienced with closure studs in the reactor coolant pumps at its Fort Calhoun facility. The corrosion wastage was attributed to boric acid attack as a result of leakage at flexitallic gasketed joints between the pump casing and pump cover. These closure studs are 3.5 inches in diameter, and are manufactured of SA 193-B7 (AISI 4140) low-alloy, high-strength steel. Accordingly, the NRC issued Information Notice No. 80-27 on June 11, 1980 to all PWR licensees about the potential for undetected boric acid corrosion wastage and emphasized the need for supplemental visual inspection of pressure-retaining bolting in pump and valve components. Subsequently, similar occurrences of corrosion wastage from borated water leakage have been identified at other PWR plants, as discussed below.

On March 10, 1982, the NRC was notified by Maine Yankee Atomic Power Company and Combustion Engineering (C-E) that during routine disassembly of a steam generator primary manway at Maine Yankee, 6 of the 20 manway closure studs failed and another 5 were found, by ultrasonic examination using specialized techniques, to be cracked. Leakage had been noted from this manway during the current operating cycle and several efforts were made to eliminate the leakage. These efforts involved increasing the joint operating compression through torquing the studs to hydrotest levels and repeatedly injecting Furmanite sealant. Normal plant operation continued until a planned maintenance outage.

8204210330

Preliminary results of a metallurgical analysis C-E performed on the affected studs have indicated that the failure mode was stress-corrosion cracking (SCC). By Information Notice No. 82-06 (issued March 12, 1982), the Office of Inspection and Enforcement notified all licensees and construction permit holders about this degradation to emphasize the increased potential for studs to fail by the joint action of stud preload, material conditions and a corrosive environment generated by the presence of primary coolant leakage. As a follow-up to the information notice, the utility established that the root cause of leakage was due to an interference contact between the gasket retainer lip and vessel cladding which prevented proper compression of the flexitallic gasket during reinstallation of the manway cover. This problem was corrected and all 20 studs were replaced. Magnetic particle and ultrasonic examinations of the studs in manways of the other two steam generators identified no other failures.

In the last several years a significant number of incidents have been reported of bolts and studs that have failed or become severely degraded because of boric acid corrosion wastage or SCC mechanisms. Preliminary results of an NRC staff review of threaded fastener experience in operating nuclear power plants have identified that specific generic actions need to be taken before the study is complete. The staff review identified 44 incidents of threaded fastener degradation since 1964. From Table 1 it can be seen that since 1977, 15 incidents related to primary coolant pressure boundary application have been recorded. These incidents involved 9 PWR plants. Of concern is that degradation and failure of such threaded fasteners constitute a potential loss of RCPB integrity and, in the extreme case, a loss-of-coolant accident could occur, should extensive fastener failures in a pressure-retaining closure not be detected.

In some instances, it has been reported that sealant compounds have been injected into bolted closures in the RCPB as a means of convenient maintenance to control leakage. A review of the limited chemical analysis available on Furmanite indicates it has a variable composition with respect to concentration of chlorine, fluorine, and sulfur which are leachable and well recognized promoters of SCC. Consequently, prolonged exposure of this sealant to leakage and high temperature conditions causing a gradual release of its potentially corrosive ions must be taken into account.

Also, certain lubricants may be formulated with molybdenum disulfide (MoS_2) which contains a significant level of sulfide constituent. Experience suggests that MoS_2 has a pronounced tendency to decompose in the presence of high temperature and moisture conditions to release sulfide which is a known promoter of SCC.

Therefore, care should be exercised in the selection and application of lubricants and injection sealants to minimize the risk of SCC from potentially corrosive ions due to the gradual breakdown and/or synergistic interaction of such materials with prolonged exposure to leakage conditions. This would be of

particular concern for fastener materials made of high-strength low-alloy steels and, austenitic and martensitic stainless steels (i.e., 304, 316, 416, 17-4 PH, etc.) which are known to be susceptible to halogen/sulfide SCC degradation.

The above concerns are further compounded by the fact that under the present ASME Code Section XI inservice inspection rules ultrasonic examination is not required on threaded fasteners in sizes 2 inches and less in diameter (e.g., Table IWB-2500-1). However, except for the reactor coolant pump stud wastage, most failures have occurred in fastener sizes 2 inches and smaller. Furthermore, experience has clearly shown that Code-specified ultrasonic testing (UT) methods are not singularly adequate to detect corrosion wastage conditions. Moreover, the present Code UT procedures are not sufficiently sensitive to detect initiation of stress corrosion cracking (SCC) but requires the use of specialized UT techniques and calibration standards based on notch reflectors simulating critical flaw parameters to enhance reliability of detection. At the present time, visual examination (e.g., IWA 2210, VT-1) appears to be the only method to detect borated water corrosion wastage or erosion-corrosion damage and may require insulation removal and/or disassembly of the component, in some cases, in order to have direct visual access to the threaded fasteners. Therefore, degradation could go undetected when there is no clear evidence of leakage in the surrounding area. Similarly, the reliability of visual examination alone is questionable in detection of SCC initiation of threaded fasteners either in-situ or removed. Accordingly, it is necessary that a combination of nondestructive examination techniques (UT, VT-1, MT, PT) be employed to the maximum extent practical to enhance detection of the degradation mechanisms discussed above.

Actions To Be Taken by PWR Facilities Holding Operating Licenses:

The scope of action items listed below is limited to the RCPB. Included are the threaded fasteners (studs or bolts) in (1) steam generator and pressurizer manway closures, (2) valve bonnets, and pump flange connections installed on lines having a nominal diameter of 6 inches or greater and (3) control rod drive (CRD) flange and pressurizer heater connections that do not have seal welds to provide leak-tight integrity. That is, CRDs having an omega seal weld design are excluded from this bulletin action. The reactor vessel head closure studs are also excluded for those PWR licensees committed to the provisions of Regulatory Guide 1.65, "Materials and Inspection for Reactor Vessel Closure Studs."

Action Item 1 is to be completed prior to the performance of the subsequent action items. Action Item 2 is to be performed within the next cycle, but no later than the completion of the next refueling outage that is initiated after 60 days from the date of this bulletin. The report requested by Action Item 3 is to be submitted within 60 days from the date of this bulletin.

1. Where procedures do not exist, develop and implement maintenance procedures for threaded fastener practices. These procedures should

include, but not limited to the following: (1) maintenance crew training of proper bolting/stud practices, tools application, specifications and requirements, (2) detensioning and retensioning practices (torque iteration), specified tolerances, and other controls for disassembly and reassembly of component closure/seal connections, (3) gasket installation and controls, and (4) retensioning methods and other measures to eliminate reactor coolant leakage during operations.

Quality assurance measures should also be established for proper selection, procurement, and application of fastener lubricants and injection sealant compounds to minimize fastener susceptibility to SCC environments.

2. Threaded fasteners of closure connections, identified in the scope of this bulletin, when opened for component inspection or maintenance shall be removed*, cleaned, and inspected per IWA-2210 and IWA-2220 of ASME Code Section XI (1974 edition or later) before being reused.

3. NRC Information Notice Nos. 80-27 and 82-06, and similar INPO (Institute of Nuclear Power Operations) correspondence (with recommendations) have been issued in regard to corrosion problems associated with bolts/studs in RCPB closures (INPO/NSAC SER 81-12). To assist the Nuclear Regulatory Commission in its ongoing review and assessment of the scope of the problem you are asked to provide the following information for closures and connections within the scope of this bulletin:

   a. Identify those bolted closures of the RCPB that have experienced leakage, particularly those locations where leakage occurred during the most recent plant operating cycle. Describe the inspections made and corrective measures taken to eliminate the problem. If the leakage was attributed to gasket failure or its design, so indicate.

   b. Identify those closures and connections, if any, where fastener lubricants and injection sealant materials have been or are being used and report on plant experience with their application particularly any instances of SCC of fasteners. Include types and composition of materials used.

4. A written report signed under oath or affirmation under provisions of Section 182a, Atomic Energy Act of 1954 as amended, shall be submitted to the Regional Administrator of the appropriate NRC Regional Office within 60 days following the completion of the outage during which Action Item 2 was performed. The report is to include:

   a. A statement that Action Item 1 has been completed.

* Fasteners "seized" or designed with interference fit, may be inspected in place.

b. Identification of the specific connections examined as required by Action Item 2.

c. The results of the examinations performed on the threaded fasteners as required by Action Item 2. If no degradation was observed for a particular connection, a statement to that effect, identification of the connection and, whether the fasteners were examined in place or removed is all that is required. If degradation was observed, the report should provide detailed information.

5. A written report signed under oath or affirmation under provisions of Section 182a, Atomic Energy Act of 1954 as amended, shall be submitted to the Regional Administrator of the appropriate NRC Regional Office within 60 days of the date of this bulletin. The report is to provide the information requested by Action Item 3.

Potential occupational exposure of personnel as a result of the above requirements should be considered in the program formulation process in an effort to maintain incurred exposures as low as reasonably achievable. Personnel exposure-savings techniques such as use of steam generator primary manway cover-handling fixtures offer substantial time and man-rem savings.

This request for information was approved by the Office of Management and Budget under clearance number 3150-0086. Comments on burden and duplication should be directed to the Office of Management and Budget, Reports Management, Room 3208, New Executive Office Building, Washington, D.C. 20503.

While no specific request or requirement is intended, the following information would be helpful to the NRC in evaluating the cost of this bulletin:

1. Staff time to perform requested inspection.
2. Radiation exposure attributed to requested inspections.
3. Staff time spent to prepare written responses.

If you have any questions regarding this matter, please contact the Regional Administrator of the appropriate NRC Regional Office, or this office.

R C DeYoung

Richard C. DeYoung, Director
Office of Inspection and Enforcement

Technical Contact: W. J. Collins
301-492-4780

Attachments:
1. Table 1
2. List of Recently Issued IE Bulletins

TABLE 1. SUMMARY OF DEGRADED THREADED FASTENERS IN REACTOR COOLANT PRESSURE BOUNDARY

Degraded Reactor Coolant Pressure Boundary Threaded Fasteners	No. of Reported Incidents	Plants (Year Incident Reported) & Reactor Vendor	Mode of Failure*
Pressurizer manway closure studs	2	Calvert Cliffs 2 (1981) C-E St. Lucie 1 (1978) C-E	BC BC
Steam generator manway closure studs	7	Maine Yankee (1982) C-E Oconee 3 (1980) B&W Arkansas 1 (1978) B&W Arkansas 1 (1980) B&W Calvert Cliffs 1 (1980) C-E St. Lucie 1 (1977) C-E San Onofre 1 (1977) W	SC SC BC SC BC BC SC
Reactor coolant pump closure studs	5	Ft. Calhoun (1980) C-E Calvert Cliffs 1 (1980) C-E Calvert Cliffs 2 (1981) C-E Oconee 3 (1981) B&W Oconee 2 (1981) B&W	BC BC BC BC BC
Safety injection check valve studs	1	Calvert Cliffs 2 (1981) C-E	BC

*SC = stress corrosion; BC = borated water corrosion.

Attachment 2
IEB 82-02
JUne 2, 1982

LIST OF RECENTLY ISSUED IE BULLETINS

Bulletin No.	Subject	Date of Issue	Issued to
82-01 Rev. 1	Alteration of Radiographs of Welds in Piping Subassemblies	05/07/82	All power reactor facilities with an OL or CP
82-01	Alteration of Radiographs of Welds in Piping Subassemblies	03/31/82	The Table 1 facilities for action and to all others for information
81-02 Supplement 1	Failure of Gate Type Valves to Close against Differential Pressure	08/18/81	All power reactor facilities with an OL or CP
81-03	Flow Blockage of Cooling Water To Safety System Components by CORBICULA SP. (ASIATIC CLAM) and MYTILUS SP. (MUSSEL)	04/10/81	All power reactor facilities with an OL or CP
81-02	Failure of Gate Type Valves to Close Against Differential Pressure	04/09/81	All power reactor facilities with an OL or CP
81-01 Rev. 1	Surveillance of Mechanical Snubbers	03/04/81	Specific power reactor facilities with a CP
80-17 Supp. 5	Failure of Control Rods to to Insert During a Scram at a BWR	02/13/81	To all specified BWRs with an OL & All BWRs with a CP
81-01	Surveillance of Mechanical Snubbers	01/27/81	All power reactor facilities with an OL and selected power reactor facilities with a CP

OL = Operating License
CP = Construction Permit

Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

June 7, 1982

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
Locked Rotor and Shaft Break
Transients
NRC Docket Nos. 50-454, 50-455,
50-456 and 50-457

Dear Mr. Denton:

This is to provide additional information regarding reactor coolant pump locked rotor and shaft break transients at Byron and Braidwood. Review of this information should close Confirmatory Issue 34 of the Byron SER.

The Byron/Braidwood FSAR documents core performance analyses of situations involving either reactor coolant pump rotor seizure or shaft break. In both cases the reactor trips in less than one-tenth of a second and the maximum clad temperature is reached in less than four seconds. The transients are essentially over in ten seconds.

The Staff has pointed out that the turbine would also trip in such a transient and a consequential interruption of power to plant auxiliaries including reactor coolant pumps could make the transient more severe. Additional analyses of such events and a re-examination of the limiting single failure were requested. As discussed below, representative transients of this type have been evaluated and the consequences determined to be acceptable.

Consequential Interruption Scenarios

Consequential interruption of power to plant auxiliaries is possible in only two ways. Switching of bus feeds at the time of generator trip could involve a breaker failure that would leave one or more reactor coolant pumps without power. Trip of the turbine could disrupt grid stability that might result in low voltage or frequency trip of reactor coolant pumps.

Switching

At Byron and Braidwood the generator trip is delayed thirty seconds past the turbine trip caused by a reactor trip in a loss of flow transient. During this delay period power flow through the main transformers reverses to maintain generator terminal voltage and supply auxiliary buses that feed reactor coolant pumps. After thirty seconds these buses are automatically switched to system auxiliary transformers fed from the 345 kv yard. This switching is delayed to provide an extra measure of turbine overspeed protection. It also precludes switching failures which could interrupt power to plant auxiliaries during the first few crucial seconds of a locked rotor or broken shaft transient.

As noted in the FSAR, the locked rotor event concurrent with a loss of offsite power at the time of generator trip has been evaluated. Without offsite power this transient ultimately results in the coastdown of the reactor coolant flow when the reactor coolant pumps are tripped. As shown by the FSAR analyses summarized earlier this will have little effect because the severity of the transient will have turned around.

Grid Stability

The turbine trip associated with a locked rotor or broken shaft transient would not cause instability on the Commonwealth Edison grid and could not complicate the transient further. From our operating experience at Zion Station, a plant similar in size, no unit trip has ever caused a noticeable instability in the grid. System voltage has never dropped to the point of tripping reactor coolant pumps.

If grid instability should ever be induced by a locked rotor or broken shaft transient, its effect on the plant would be slower than the initiating transient itself. The grid instability would manifest itself in a reduced frequency. High fault currents would then occur in the transmission lines. Automatic breaker trips would begin in a cascading manner to isolate the faults. Since the highest fault currents would be seen first at points furthest from the station, the station would be the last, or close to the last, point to be isolated from the grid. This series of events would take longer than four seconds, which is the time of the peak of the transient, and neither the frequency nor voltage decay to the point of pump trip during this time. The analysis of the locked rotor event is applicable to the broken shaft event with concurrent loss of offsite power.

Single Failure

The Staff has also suggested that these events should be analyzed assuming the worst single failure of a safety system active component.

The single active failure assumed in the Locked Rotor/Pump Shaft Break analysis presented in the Byron/Braidwood FSAR is the loss of one protection train used to initiate a reactor trip on a low flow signal. This failure, or any other failure, will not have an effect on the transient because of the transient's relatively short duration. In less than four seconds after initiation of the accident, the safety parameters of concern (departure from nucleate boiling ratio, reactor coolant pressure and clad temperature) have reached their maximum (or minimum for DNBR) values and begin to approach steady state values.

The following are additional single active failures that are assumed in other accidents. As noted, none of them will increase the severity of the Locked Rotor accident.

- Loss of a safety injection train - Safety injection is not required to mitigate the effects of this accident.
- Loss of an auxiliary feedwater pump - The Locked Rotor Accident is turned around before the auxiliary feedwater (AFW) pumps could be turned on.
- PORV stuck open - The Locked Rotor Accident is turned around before the stuck open PORV would have any effect. A stuck open PORV is analyzed in Section 15.6.1.
- Failure of a main steam isolation valve (MSIV) - The MSIVs are not required to mitigate the effects of the Locked Rotor Accident.
- Failure of a feedline isolation valve - The Locked Rotor Accident is turned around before this failure.
- Stuck open secondary side valve - Radiological effects due to a stuck open valve on the secondary side are of no consequence for the following reasons. Steam releases assumed in the radiological analysis for the locked rotor accident assumed a conservatively high steam release for eight hours. The conservatisms include (1) no steam dump to the condenser (2) a high rate of decay heat and (3) a high level of energy stored in the reactor system structure.

  Steam releases computed on the above basis will always exceed releases that could occur while bringing the plant to a shutdown condition after the accident. Therefore, a stuck open relief, safety, or dump valve is of no consequence to the radiological releases calculated.

Based on the short duration of the transient and a review of other single failures, the loss of a protection train is an appropriate single failure for the locked rotor accident both with and without offsite power available.

In summary, the generator trip delay and grid stability provide assurance that the events analyzed in the FSAR are appropriate design basis loss of flow transients. Failure of one protection train is an appropriate design basis failure for the transient.

Please direct further questions regarding this matter to this office.

One signed original and fifteen copies of this letter are provided for your use.

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Adminstrator

lm

4195N

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

JUN 14 REC'D

JUN 1 0 1982

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

Mr. Louis O. DelGeorge
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. DelGeorge:

Subject: Loose Parts Monitoring Program for Byron/Braidwood Stations

In Amendment 37 to the Byron/Braidwood Final Safety Analysis Report (FSAR), Commonwealth Edison committed to providing a comprehensive report on the loose parts monitoring program for the Byron/Braidwood Stations. This final design report would demonstrate that the loose parts monitoring system (LPMS) was substantially in conformance with Regulatory Guide 1.133, Revision 1, "Loose-Parts Detection Program for the Primary System of Light-Water-Cooled Reactors," issued in May 1981.

Recently, the staff had our consultant, Oak Ridge National Laboratory (ORNL), review the LPMS information submitted in the Byron/Braidwood FSAR. The attached table contains the results of this review and lists a number of areas where your submittal has provided insufficient information. The purpose of this letter is to request that Commonwealth Edison address these concerns in its final design report. Including this information, along with the information already provided and the results of the preoperational tests, will ensure that a complete and comprehensive LPMS report is provided by Commonwealth Edison.

If you should require any additional information please contact the project managers Mr. Steve Chesnut for the Byron Station or Mr. Kenneth Kiper for the Braidwood Station.

Sincerely,

B J Youngblood

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

cc: See next page

Mr. Louis O. DelGeorge
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

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

Paul M. Murphy, Esq.
Isham, Lincoln & Beale
One First National Plaza
42nd Floor
Chicago, Illinois 60603

C. Allen Bock, Esq.
Post Office Box 342
Urbana, Illinois 61801

Thomas J. Gordon, Esq.
Waaler, Evans & Gordon
2503 S. Neil
Champaign, Illinois 61820

Ms. Bridget Little Rorem
Appleseed Coordinator
117 North Linden Street
Essex, Illinois 60935

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

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
RR#1, Box 79
Braceville, Illinois 60407

Mr. James G. Keppler
U. S. NRC, Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. Louis O. DelGeorge
Driector of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

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

Paul M. Murphy, Esq.
Isham, Lincoln & Beale
One First National Plaza
42nd Floor
Chicago, Illinois 60603

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

Ms. Bridget Little Rorem
Appleseed Coordinator
117 North Linden Street
Essex, Illinois 60935

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

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

Myron Cherry, Esq.
Cherry & Flynn
Suite 3700
Three First National Plaza
Chicago, Illinois 60602

Mr. William Fourney
U. S. Nuclear Regulatory Commission
Byron/Resident Inspectors Office
4448 German Church Road
Byron, Illinois 61010

Ms. Diane Chavez
602 Oak Street, Apt. #4
Rockford, Illinois 61108

Mr. James G. Keppler
U. S. NRC, Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Enclosure 1

DETAILED COMMENTS

We have reviewed the subject LPMS descriptions with regard to specific areas of nonconformance to RG 1.133, and in accordance with your guidance, we are attempting to keep our responses concise by means of the accompanying table, which is keyed to Section C (Regulatory Position) of the RG.

In addition to the table, it may be appropriate to identify and discuss some particular areas of deficiency that we feel are especially important to the establishment of effective loose-part detection programs.

1. Periodic data review and long-term maintenance and calibration

Our previous survey of LPMS implementation by the industry revealed that once a system is installed, calibrated, and placed in initial operation by the LPMS supplier it is typically neglected by the plant operating personnel with regard to its continued operation, maintenance, assurance of calibrations, and monitoring of output. In this context, we observe that all the submittals under review failed to state how the licensees plan to address these important areas, as defined in Sect. 3.a of RG 1.133.

2. Alert logic and consistency with recommended detection sensitivity

Two of the submittals under review (Watts Bar and Midland) describe a method to reduce false alarms that is based on self-adjusting threshold logic which "tracks" varying acoustic background noise at different plant operating conditions. Properly implemented, this feature is acceptable (see RG 1.133 Sects. 2.d and 3.a(2)(e)), but the proposed calibration and verification of required system sensitivity at plant shutdown conditions implies to us that, under normal operating (higher background noise) conditions, the 0.5 ft-lb sensitivity requirement may not be met or, worse yet, the system may automatically be rendered insensitive to the detection of safety-significant loose parts without the knowledge of the plant operators.

3. Reference to technical specification

While RG 1.133 Sect. 4.f requires the SAR to make reference to a Tech Spec which defines a limiting condition for operation with regard to inoperability of the LPMS, none of the submittals under review addressed this subject. This again illustrates an apparent lack of commitment to a meaningful loose-parts program on the part of the licensees.

SUMMARY OF REVIEW OF LPMS DESCRIPTIONS

RG 1.133 Section	Byron/ Braidwood
C.1 System Characteristics	
a. Two sensors at each natural collection region	C **
b. Sensitivity of 0.5 ft-lb within 3 ft of sensor	NI
c. Physical separation of instrumentation channels	NI
d. Automatic data acquisition (tape recorder)	C
e. Automatic comparison of signal to an alert level	C
f. Periodic system operational verification and calibration	NI
g. Ability to function after seismic event	C
h. Quality of system components	C
i. Ease of repair to minimize radiation exposure	NI

** Symbols explained in KEY on final page

RG 1.133 Section	Byron/ Braidwood
C.2. Establishing the Alert Level	
a. Logic to recognize LP in presence of noise	I
b. Override of noise caused by control rod movement, etc.	NI
c. Alert level a function of plant operating conditions	NI
d. Compensation for different background noise on sensors	C
C.3. Using the Data Acquisition Modes	
a. Manual Mode	
(1) Pre-op tests to establish alert level	C
(2) Startup and power operation	
a. Submit alert level within 90 days after startup	NA
b. Perform channel check each 24 hours	NI
c. Listen to audio output each 7 days	C
d. Perform functional test each 31 days	NI
e. Verify background noise each 92 days	I

RG 1.133 Section	Byron/ Braidwood
(3) Verify channel calibration each 18 months	NI
b. Automatic data recording when alert level is exceeded	C
C.4. Content of Safety Analysis Reports	
a. Sensor type, location, mounting, and criteria for these	I
b. Description of data acquisition, recording, and calibration	I
c. Major sources of extraneous noise	C
d. Quality assurance of data	NI
e. Description of alert level determination and alert logic	I
f. Reference to technical specification	NI
g. Description of diagnostic procedures used to confirm loose part	NI
h. Channel check procedures	NI
i. Maintenance procedures to minimize radiation exposure	NI
j. Training program	C
k. Verification that LPMS will function after a seismic event	C

RG 1.133 Section	Watts Bar	Midland	Byron/ Braidwood
C.5. Technical Specification for Loose-Part Detection System	NA	NA	NA
C.6. Notification of a Loose Part	NA	NA	NA

KEY: C - Conformance with RG 1.133

NC - Nonconformance with RG 1.133

I - Insufficient information provided

NI - No information provided

NA - Not applicable at this time

Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

June 14, 1982

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
Safe Shutdown Analysis
NRC Docket Nos. 50-454, 50-455
50-456 and 50-457

Reference (a): November 17, 1981 letter from
T. R. Tramm to H. R. Denton.

Dear Mr. Denton:

This is to provide information regarding the ability of the Byron and Braidwood units to be safely shutdown in the event of a fire. The first report of our study was provided in reference (a).

The first report identified problem zones for which further analysis was required. The revised report contained in Attachment A to this letter addresses those zones. In each case cables have been rerouted or a more detailed analysis has been performed to justify the present design. The revised report also includes neutron monitoring equipment and the associated cables. Incore thermocouple cabling will be addressed in a future revision to this report.

The attached report will be incorporated into the Byron/Braidwood Fire Protection Report in an update to that report which is to be submitted on August 2, 1982.

Please direct any questions regarding this report to this office.

One signed original and fifteen copies of this letter and the attachment are provided for your review.

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator

lm

4331N

ATTACHMENT A

Byron/Braidwood Safe Shutdown Analysis

4331N

SSINS No.: 6835
IN 82-17

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

JUN 15

June 11, 1982

IE INFORMATION NOTICE NO. 82-17: OVERPRESSURIZATION OF REACTOR COOLANT SYSTEM

Addressees:

All nuclear power reactor facilities holding an operating license (OL) or construction permit (CP).

Purpose:

This information notice is provided as a notification of two events that may have safety significance. It is expected that recipients will review the information for applicability to their facilities. No specific action or response is required.

Description of Circumstances:

On November 30, 1981, Florida Power and Light Company reported that the Turkey Point Unit 4 reactor coolant system (RCS) was overpressurized on November 28 and 29 during startup following a refueling outage. The reactor was shut down and the RCS was in a water solid condition with a pressure and temperature of approximately 310 psig and 110°F, respectively. Two separate transients that resulted in overpressure conditions of 1100 and 750 psig at 110°F occurred for which the overpressure mitigating system (OMS) failed to operate. These events exceeded the pressure limit of 480 psig at 110°F specified in Technical Specifications which prescribe the allowable pressure and temperature limits to prevent reactor vessel brittle fracture.

The OMS is specifically designed to prevent this type of overpressurization. The OMS did not operate as designed because:

1. After the first event a pressure transmitter isolation valve was found closed and was opened. This transmitter provides input into the OMS circuit to automatically open a power operated relief valve (PORV) if the reactor coolant system exceeds the allowable pressure for RCS temperature;

2. The summator failed in the electrical circuitry which prescribes the pressure at which the OMS is to initiate PORV actuation. The failed summator was identified and corrected after the second event. The OMS surveillance procedure in use before the event did not include testing the summator, in that the test signal bypassed the summator; and

3. The redundant OMS circuit was out of service for calibration.

Before each event the reactor coolant system inventory was being maintained by charging from the chemical and volume control system and letdown through the residual heat removal (RHR) system. Each event was initiated with a pressure spike caused by the start of a reactor coolant pump which resulted in isolation of letdown by automatic closure of the RHR system isolation valve. During both occurrences, the operator took immediate action to stop the charging pump which was providing the source of rapid pressurization. Within two minutes the operator decreased pressure to the desired level by manually opening the PORV and securing the pressurizer heaters in addition to securing the charging flow. Timely operator action to completely prevent the overpressurization was precluded by the rapidity of the transient.

Following the events, OMS surveillance procedures were revised to include testing of the summator. Other procedural changes include additional equipment checks to ensure OMS operability. Westinghouse performed a fracture mechanics analysis based on the method of Appendix G, Section III of the ASME Boiler and Pressure Vessel Code. The analysis showed that these transients had neither impaired the integrity of the reactor vessel, nor significantly affected the fatigue life of the vessel. A Florida Power and Light Company consultant reviewed the analysis and concurred with the Westinghouse conclusion.

In a separate event, on May 23, 1982 Virginia Electric Power Company (VEPCO) reported that the overpressure protection system (OPS) at North Anna was inoperable from May 19-22. The OPS had not been called upon to operate during this time. The reactor was in cold shutdown and for two days the reactor coolant system was in the water solid condition. Initially, one OPS system was declared inoperable when the pressure in the "A" nitrogen supply reservoir dropped below the minimum pressure required to maintain the PORV operable. Two days later, a low pressure alarm occurred on the "B" nitrogen supply reservoir. An isolation valve between the reservoir and the "B" OPS system had been closed for an indeterminant period (possibly as long as eight days), isolating the nitrogen supply to the "B" PORV. Therefore both PORVs were inoperable. Initial investigation discovered that system procedure did not include OPS valve lineups, and the incorrectly positioned valve was not shown on plant drawings in use at that time. VEPCO is presently taking action to correct these problems.

Each of the above events involved failure of two redundant systems designed to provide overpressure protection. The concern is that without prompt operator action such failures increase the potential for brittle fracture of the reactor pressure vessel from overstress during pressure transients.

If you have any questions regarding this matter, please contact the Regional Administrator of the appropriate NRC Regional Office, or this office.

for Robert L. Baer
Edward L. Jordan, Director
Division of Engineering and
Quality Assurance
Office of Inspection and Enforcement

Technical Contact: R. A. Holland
301-492-4791

W. R. Mills
301-492-4791

Attachment:
List of Recently Issued IE Information Notices

Attachment
IN 82-17
June 11, 1982

LIST OF RECENTLY ISSUED
IE INFORMATION NOTICES

Information Notice No.	Subject	Date of Issue	Issued to
82-16	HPCI/RCIC High Steam Flow Setpoints	5/28/82	All power reactor facilities holding an OL or CP
82-15	Notification of the Nuclear Regulatory Commission (NRC)	5/28/82	All NRC licensees and all power reactor facilities holding a CP
82-14	TMI-1 Steam Generator/Reactor Coolant System Chemistry/ Corrosion Problem	5/12/82	All power reactor facilities holding an OL or CP
82-13	Failures of General Electric Type HFA Relays	5/10/82	All power reactor facilities holding an OL or CP
82-12	Surveillance of Hydraulic Snubbers	4/21/82	All power reactor facilities holding an OL or CP
82-11	Potential Inaccuracies in Wide Range Pressure Instruments used in Westinghouse Designed Plants	04/09/82	All power reactor facilities holding an OL or CP
82-10	Following up Symptomatic Repairs to Assure Resolution of the Problem	04/09/82	All power reactor facilities holding an OL or CP
82-09	Cracking in Piping of Makeup Coolant Lines at B&W Plants	03/31/82	All power reactor facilities holding an OL or CP
82-08	Check Valve Failures on Diesel Generator Engine Cooling System	03/26/82	All power reactor facilities holding an OL or CP
82-07	Inadequate Security Screening Programs	03/16/82	All power reactor facilities holding an OL or CP

OL = Operating License
CP = Construction Permit