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

Docket Nos. 50-456
50-457

AMENDMENT TO INDEMNITY AGREEMENT NO. B-102
AMENDMENT NO. 6

Effective May 20, 1988, Indemnity Agreement No. B-102, between Commonwealth Edison Company and the Nuclear Regulatory Commission, dated October 8, 1985, as amended, is hereby further amended as follows:

Item 3 of the Attachment to the indemnity agreement is deleted in its entirety and the following substituted therefor:

Item 3 - License number or numbers

SNM-1938	(From 12:01 a.m., October 8, 1985, to 12 midnight, October 16, 1986, inclusive)
SNM-1945	(From 12:01 a.m., July 27, 1987, to 12 midnight, December 17, 1987, inclusive)
NPF-59	(From 12:01 a.m., October 17, 1986, to 12 midnight, May 20, 1987, inclusive)
NPF-70	(From 12:01 a.m., May 21, 1987, to 12 midnight, July 1, 1987 inclusive)
NPF-72	(From 12:01 a.m., July 2, 1987)
NPF-75	(From 12:01 a.m., December 18, 1987, to 12 midnight, May 19, 1988 inclusive)

NPF-77 (From 12:01 a.m., May 20, 1988)

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cecil O. Thomas

Cecil O. Thomas, Chief
Policy Development and Technical
Support Branch
Program Management, Policy Development
and Analysis Staff
Office of Nuclear Reactor Regulation

Accepted ____________________, 1988

By ____________________
COMMONWEALTH EDISON COMPANY

NUREG-1002
Supplement No. 6

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

Docket Nos. STN 50-456 and STN 50-457

Commonwealth Edison Company

U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

May 1988

ABSTRACT

In November 1983, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-1002) regarding the application filed by the Commonwealth Edison Company, as applicant and owner, for a license to operate Braidwood Station, Units 1 and 2 (Docket Nos. 50-456 and 50-457). The first supplement to NUREG-1002 was issued in September 1986; the second supplement was issued in October 1986; the third supplement was issued in May 1987; the fourth supplement was issued in July 1987 in support of the full power license for Unit 1; the fifth supplement was issued in December 1987 in support of the low power license for Unit 2. This sixth supplement to NUREG-1002 is in support of the full-power license for Unit 2 and provides the status of items that remained unresolved at the time Supplement 5 was published. The facility is located in Reed Township, Will County, Illinois.

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APPENDICES

1 INTRODUCTION AND GENERAL DESCRIPTION OF FACILITY

1.1 Introduction

In November 1983, the Nuclear Regulatory Commission (NRC) staff issued its Safety Evaluation Report (SER) (NUREG-1002) on the application filed by the Commonwealth Edison Company, as applicant and owner, for a license to operate Braidwood Station, Units 1 and 2 (Docket Nos. 50-456 and 50-457). At that time, the staff identified items that had not been resolved with the applicant. The first supplement to NUREG-1002 was issued in September 1986; the second supplement to NUREG-1002 was issued in October 1986; the third supplement to NUREG-1002 was issued in May 1987; the fourth supplement was issued in July 1987. The fifth supplement to the SER provided the staff's evaluation of the open items that have been resolved in support of the low-power license. This sixth supplement to the SER provides the staff's evaluation of the open items that have been resolved to date or addresses changes to the SER that resulted from the receipt of additional information from Commonwealth Edison Company (licensee); in addition, this supplement supports the issuance of the full-power license for Unit 2.

Each section or appendix that follows is numbered the same as the corresponding SER section or appendix that is being updated. Each section is supplementary to and not in lieu of the discussion in the SER unless otherwise noted. Appendix A continues the chronology of the staff's actions related to the processing of the application for Braidwood Units 1 and 2. Appendix F lists principal staff members who contributed to this supplement. Appendix B provides the references in support of this SER.

Copies of this SER supplement are available for inspection at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Wilmington Township Public Library, 201 South Kankakee Street, Wilmington, Illinois 60481.

The NRC Project Manager for Braidwood Station, Units 1 and 2, is Mr. Stephen P. Sands. Mr. Sands may be contacted by calling (301) 492-1396 or writing:

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1.7 Summary of Outstanding Items

The current status of the outstanding items listed in the SER follows:

Part A Items	Status	Section
(1) Pump and valve operability	Closed in Supplement 2	3.9.3.2*
(2) Seismic and dynamic qualification of equipment	Closed in Supplement 2	3.10*
(3) Environmental qualification of electrical and mechanical equipment	Closed in Supplement 2	3.11*
(4) Containment pressure boundary components	Closed in Supplement 1	6.2.7
(5) Organizational structure	Closed in Supplement 1	13.1, 13.4
(6) Emergency preparedness plans and facilities	Closed in Supplement 1	13.3*
(7) Procedures generation package (PGP)	Closed in Supplement 2	13.5.2
(8) Control room human factors review	Closed in Supplement 4	18.2*
(9) Safety parameter display system	Closed in Supplement 4	18.3*
(10) Control room habitability	Closed in Supplement 3	6.4
Part B Items		
(1) Turbine missile evaluation	Closed in Supplement 1	3.5.1.3
(2) Improved thermal design procedures	Closed in Supplement 1	4.4.1
(3) TMI Action Item II.F.2: Inadequate Core Cooling Instrumentation	Closed in Supplement 1	4.4.7
(4) Steam generator flow-induced vibrations	Closed in Supplement 1	5.4.2
(5) Conformance of ESF filter system to RG 1.52	Closed in Supplement 2	6.5.1
(6) Fire protection program	Closed in Supplement 3	9.5.1

*This section includes both site-specific-related information and duplicate-plant design features.

Part B Items (Continued)	Status	Section
(7) Volume reduction system	Closed in Supplement 2	11.1, 11.4.2

1.8 Confirmatory Issues

The current status of the confirmatory issues follows:

Part A Items

(1) Applicant compliance with the Commission's regulations	Closed in Supplement 2	1.1, 3.1*
(2) Site drainage	Closed in Supplement 1	2.4.3.3
(3) Piping vibration test program	Closed in Supplement 1	3.9.2.1*
(4) Preservice inspection program	Closed in Supplement 2	5.2.4, 6.6*
(5) Reactor vessel materials	Closed in Supplement 1	5.3
(6) Electrical distribution system voltage verification	Closed in Supplement 1	8.2.4*
(7) Independence of redundant electrical safety equipment	Closed in Supplement 1	8.4.4
(8) RPM qualifications	Closed in Supplement 1	12.5
(9) Revision to Physical Security Plan	Closed in Supplement 1	13.6
(10) Control room human factors review	Opened in Supplement 4	18.2*
(11) Safety parameter display system	Opened in Supplement 4	18.3*

Part B Items

(1) Inservice testing of pumps and valves	Partially closed in Supplement 2	3.9.6

*This section includes both site-specific-related information and duplicate-plant design features.

Part B Items (Continued)	Status	Section
(2) Steam generator tube surveillance	Closed in Supplement 1	5.4.2.2
(3) Charging pump deadheading	Closed in Supplement 1	6.3.2, 7.3.2
(4) Minimum containment pressure analysis for performance capabilities of ECCS	Closed in Supplement 1	6.2.1.5
(5) Containment sump screen	Closed in Supplement 1	6.2.2
(6) Containment leakage testing vent and drain provisions	Closed in Supplement 1	6.2.6
(7) Confirmatory test for sump design	Closed in Supplement 1	6.3.4.1
(8) IE Bulletin 80-06	Closed in Supplement 1	7.3.2.2
(9) Remote shutdown capability	Closed in Supplement 2	7.4.2.2
(10) TMI Action Plan Item II.D.1	Partially closed in Supplement 1	3.9.3.3, 5.2.2
TMI Action Plan Item II.K.3.1	Closed in Supplement 1	7.6.2.7
TMI Action Plan Item III.D.1.1	Closed in Supplement 1	9.3.5
(11) SWS process control program	Closed in Supplement 2	11.4.1
(12) Noble gas monitor	Closed in Supplement 2	11.5.2
(13) RCP rotor seizure and shaft break	Closed in Supplement 1	15.3.6
(14) Anticipated transients without scram (ATWS)	Partially closed in Supplement 2	15.6
(15) Evaluation of compliance with 10 CFR 50.55a(a)(3)	Closed in Supplement 2	5.2.4.4
(16) Steam generator tube failure	Opened in Supplement 1	15.4.3

1.9 License Conditions

The current status of the license conditions follows:

Part A Items	Status	Section
(1) Inservice inspection program	Closed in Supplement 3	5.2.4, 6.6*
(2) Natural circulation testing	Closed in Supplement 1	5.4.3*
(3) Response time testing	Closed in Supplement 1	7.2.2.5*
(4) Steam valve inservice inspection	Closed in Supplement 1	10.2*
(5) Implementation of secondary water chemistry monitoring and control program as proposed by the Byron/Braidwood FSAR	Closed in Supplement 1	10.3.3*
(6) TMI Item II.F.1: Iodine/Particulate Sampling	Closed in Supplement 3	11.5.2
Part B Items		
(1) Masonry walls	Closed in Supplement 2	3.8.3
(2) TMI Item II.B.3 postaccident sampling	Closed in Supplement 1	9.3.2
(3) Fire protection program	Open	9.5.1
(4) Emergency diesel engine auxiliary support systems	Closed in Supplement 3	9.5.4.1

*This section includes both site-specific-related information and duplicate-plant design features.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

By letter dated March 29, 1988, as amended with supplemental information in a letter dated April 11, 1988, the licensee has provided information related to a concern about contaminated grease in Limitorque motor operated valves. This information was discussed at Braidwood Station with members of the staff on April 4, 1988. Additional information was provided by letters dated April 21 and 22, 1988.

Exxon Nebula EP-0 and EP-1 calcium greases have been environmentally qualified by Limitorque for use in safety-related Limitorque operators located inside containment at Braidwood Units 1 and 2. Sun 50 EP lithium lead lubricant has been qualified by Limitorque for non-EQ use in safety-related operators located outside containment. In response to reports of mixed greases in Limitorque main gear cases, the licensee performed a 100% grease sampling program on all 263 Braidwood Unit 1 and 2 operators in 1985. It was determined by evaluation of visual and chemical analysis results that 103 Limitorque operators contained grease which was unacceptable (mixed or contaminated grease and any grease other than Nebula EP-0 and EP-1 for EQ applications). All of these operators were disassembled, parts and operator housing degreased with VARSOL #3 solvent, inspected for acceptable cleanliness, regreased with Exxon Nebula EP-0 or EP-1 and reassembled. Exxon Nebula EP-1 was predominantly used for regreasing. This Limitorque grease sampling and change-out program occurred from late 1985 to early 1987. The 1985 samples may not have been representative since the samples were obtained from the most convenient grease plug, usually at the top.

Recently, samples of grease were obtained from ten Limitorque operators to verify the quality of the grease. One of the 10 samples indicated the presence of a mixed grease. As a result of the observation, all 263 safety-related Limitorque operators from Braidwood Units 1 and 2 were visually inspected, sampled, and evaluated for mixed or contaminated grease. The 1988 sampling program involved inserting a nylon tie wrap through each accessible grease plug, (top, middle, and bottom of operator, where possible) and twisting the tie wrap to collect a representative grease sample.

Evaluation of visual inspection and sampling analysis showed the grease was unacceptable (visual examination indicating mixed grease or chemical analysis showing $\geq$ 5% contaminant) in 68 Limitorque operators. Those unacceptable motor operators have been cleaned and regreased in Unit 1 and are being regreased prior to entering Mode 2 operation in Unit 2.

Sampling results for 132 additional operators indicated that the grease was acceptable with no further remedial action required. The licensee's acceptance criteria are: 1) satisfactory sensory tests including no identifiable mixture of grease products and 2) chemical test indicating that the primary grease exceeds the secondary or contaminant grease by a ratio equal to or greater than 50 to 1 ($\leq$ 2% contaminant).

The visual and chemical analysis of the remaining 63 operators (33 in Unit 1 and 30 in Unit 2) showed an identifiable discoloration (indication of mixed grease) or a chemical test with the ratio of the primary grease to the secondary grease of between 50 to 1 and 20 to 1 (2 to 5% contaminant) and were subject to penetration testing. Mixing of greases can cause a compatibility problem due to deleterious chemical reactions resulting in either a hardening or softening of the grease. The standard penetration test (ASTM D217, Reference 2) indicates that a mixed grease is compatible if the worked penetration does not change more than 30 points (Reference 1). The licensee has committed to change-out the grease in the 2-5% contaminant range in Braidwood Unit 2 operators prior to entering mode 2. The basis for the acceptability of the mixed grease at ratios of between 50 to 1 and 20 to 1 in the 33 Unit 1 operators is a penetration test range of 295 to 400 points. This acceptance band is defined as a penetration test result that is ±30 points of the midpoint of the range for the two qualified Limitorque operator greases (Nebula EP-0, penetration range 355-385 and Nebula EP-1, penetration range 310-340).

The licensee has committed to disassemble, degrease, inspect for cleanliness, regrease with Exxon Nebula EP-1, and reassemble the Unit 1 operators with 2 to 5% contaminant during the next refueling outage, presently scheduled to occur in May of 1989.

We have evaluated the information provided by the licensee in order to answer two concerns. The first being whether the mixed greases are compatible and will retain lubricating characteristics during normal operation. The second concern involves mixed grease lubricating characteristics following environmental exposure to a Design Basis Accident to permit valve operability.

In addressing the first concern, we note that most mixed greases that contain more than 20 to 1 calcium base to lithium base grease have passed the penetration test. This test measures the consistency of the grease and is used in determining the National Lubricating Grease Institute (NLGI) grade classification (Exxon Nebula EP-0 has a NLGI Grade No. 0 with a penetration of 355-385 and Exxon Nebula EP-1 has a NLGI Grade No. 1 with a penetration of 310-340 points). Since the penetration tests met the acceptance criteria for Exxon Nebula EP-0 and EP-1 of 310-385 ±15 points (295-400), there were no apparent chemical interactions of the mixed greases to significantly change its consistency. Furthermore, the contaminated greases have been mixed by stroking the Limitorque operator about 26 times. This provides assurance that the greases are sufficiently mixed to complete any chemical reactions that may occur between the two grease constituents.

During the stroking of these operators, no valve operability problems attributable to grease were identified. We, therefore, conclude that those operators which have either grease with an identifiable discoloration or with Exxon Nebula EP-0 or EP-1 contaminated with less than 5% Sun EP 50, will operate during normal operation without impeding operator performance.

The second concern involves mixed grease lubricating characteristics following environmental exposure of the valve operators to a Design Basis Accident. Exxon Nebula EP-0 and EP-1 grease have been environmentally qualified in Limitorque operators at 2×10^8 Rads radiation dose, and 140°F ambient and 340°F transient temperatures. Sun 50 EP grease has been qualified for less harsh environmental conditions for outside containment Limitorque operators (2×10^7 Rads).

There is limited data on the effects of radiation exposure on mixed greases. Niagara Mohawk environmental testing of a mixture of calcium and lithium lead based greases resulted in no unacceptable degradation due to radiation. Although the test data is not directly applicable to the Braidwood grease mixture (similar grease bases, but unidentified trade names and composition), it does provide an indication of the radiation resistance of a mixture of similar chemical constituents.

Of the 33 operators in Braidwood Unit 1 with 2 to 5% contaminant in the grease, 29 will receive a radiation exposure of $\leq 1 \times 10^7$ Rads after an accident. At this radiation exposure the contaminated grease is not expected to degrade. Confirmation of no expected accident radiation degradation of the grease mixture in the four remaining operators ($> 10^7$ Rads) is needed. These operators are:

inside containment	Radiation, Rads
1OG057A Hydrogen Recombiner	2×10^8
1RY8000A Pressurizer PORV	2×10^8
outside containment	
1CV112B VCT Outlet	8×10^7
1CV112C VCT Outlet	8×10^7

The licensee committed that environmental test data will be obtained on the 2 to 5% grease assuring that there will be no unacceptable degradation of lubricating properties at radiation exposures up to 2×10^8 Rads.

Based on the above evaluation we have concluded the following:

- Qualified grease with 2% or less of contaminant can be used in Braidwood Unit 1 and 2 operators inside containment since the grease mixture is compatible and will withstand the post-accident environment. Prior to entering mode 2 in Braidwood Unit 2 the completion of the regreasing of operators with 2 to 5% contaminant must be confirmed.

- Grease with 2 to 5% contaminant can be used in Braidwood Unit 1 in the interim based on the commitment that the licensee will disassemble, degrease, inspect for cleanliness, regrease with Exxon Nebula EP-1, and reassemble these operators during the next refueling outage, presently scheduled to occur in May 1989.

- Confirmation of no expected degradation in a radiation environment of up to 2×10^8 Rads is required for the four operators which can receive an exposure greater than 10^7 Rads. The licensee has committed to perform radiation tests on qualified grease with 2 to 5% contaminant within 60 days of entering mode 2 at Braidwood Unit 1 to provide assurance that there will be no degradation of lubricating properties at radiation exposures up to 2×10^8 Rad.

3.11 Environmental Qualification of Safety-Related Electrical Equipment

During a Nuclear Regulatory Commission (NRC) Environmental Qualification (EQ) inspection and audit conducted at the Braidwood Unit 2 facility during February and March of 1988, it was determined that the environmental qualification of four Bunker Ramo supplied containment penetration assemblies which protect post-accident monitoring instrumentation circuits had not been adequately demonstrated. Further review by the NRC staff led them to conclude that the tests used by the licensee to environmentally qualify the assemblies per the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Part 50.49(f) were inadequate with respect to licensing criteria applicable to the facility and did not demonstrate environmental qualification of the assemblies. The NRC communicated this position to the licensee in a letter dated April 8, 1988.

On April 7, 1988, the licensee submitted a request for a "conditional schedular exemption" from the requirements of 10 CFR 50.49 regarding the environmental qualification of the Bunker Ramo containment penetration assemblies to preclude delay in issuance of the full power license for the facility. On April 15, 1988, the licensee submitted a follow-up request for a temporary exemption from the requirements of 10 CFR Part 50.49(j) until startup following the surveillance outage in January 1989. Along with this request, the licensee indicated that prior to startup from the January outage, the unqualified Bunker Ramo containment penetration assemblies in question will either be qualified or replaced with ones which have been previously demonstrated to be qualified per the licensing criteria applicable to the facility. In conjunction with this latter request, the licensee submitted a detailed safety evaluation in support of their conclusion that the temporary exemption would not present an undue risk to public health and safety. On April 29, 1988, the licensee provided additional technical information in response to NRC staff concerns regarding the environmental qualification of the assemblies. The staff met with the licensee on May 2 and 3, 1988 to discuss the information provided in support of the exemption request.

The purpose of the staff's evaluation was to determine whether or not there exists a technical basis on which to grant the temporary exemption requested by the licensee. In performing this evaluation, the staff has considered information provided by the licensee and other independent sources.

The requirements related to equipment qualification are contained in 10 CFR 50.49. Additional information is contained in NUREG-0588, Rev. 1, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment. The principal technical issue regarding the Bunker Ramo containment electrical penetration assemblies is the degree to which the assemblies would experience degradation of insulation resistance (IR) in instrumentation circuits during accidents which create a harsh environment. Reductions in IR can lead to current leakage which can bias instrument output causing erroneous readings and, if high enough, cause the instrument to fail. Analyses performed at Sandia National Laboratory indicate that biasing begins to become significant (i.e., greater than a few percent) in typical transmitter circuits if IR values fall below 500 kohms and, in typical resistance temperature detector (RTD) circuits, below about 50 kohms. These analyses are documented in NUREG/CR-3691. The licensee's acceptance criteria for its penetrations is an IR greater than 1000 Kohms. The Bunker Ramo test results submitted by the licensee contained numerous readings below 500 kohms with some readings indicating no resistance to ground.

Concern regarding the Bunker Ramo containment penetration assembly stems from the results of an equipment qualification inspection during which test results were reviewed showing IR measurements of less than 5 kohms. The licensee asserts that terminal blocks installed in the tested penetration circuit caused the low IR readings. The design of the circuit was modified before installation at Braidwood 2 to eliminate terminal blocks. This modification replaced the terminal blocks with a splice which has been environmentally qualified. The licensee has not confirmed the validity of this theory with a test of the Bunker Ramo penetration assemblies installed at Braidwood Unit 2.

However, in lieu of a satisfactory test for the Braidwood assembly, the licensee has referenced the results of tests on two specific major components which are part of the Bunker Ramo penetration assemblies installed at Braidwood Unit 2, and four tests on assemblies somewhat similar to the Braidwood assemblies.

As additional support for the exemption request, the licensee provided an assessment of the impact that failures of the Bunker Ramo assemblies would have on accident sequences involving harsh environments. Details regarding the assessment were provided to the NRC staff with letters dated April 7, 1988, April 15, 1988 and May 5, 1988; and in a meeting with the staff held on May 2, 1988. In performing the assessment, the licensee (1) identified the safety-related instrumentation circuits that could become degraded or fail at a Bunker Ramo containment penetration, (2) determined the availability of alternate instrument signals which would provide necessary inputs to the reactor protection and engineered safety features actuation systems (RPS and ESFAS) during applicable accident sequences, (3) assessed the adequacy of available post-accident monitoring equipment referenced in the emergency operating procedures (EOP) for bringing the plant safely to a safe shutdown condition. Item (3) was accomplished through exercises with plant operators on the Byron/Braidwood plant simulator during which the simulator was programmed to simulate instrument failures due to high leakage current at a Bunker Ramo containment penetration assembly.

The results of the licensee's assessment are the following:

(1) For purposes of accomplishing automatic safety functions of the RPS and ESFAS during accidents involving harsh environments, the containment pressure and/or steamline pressure instrument channels would be available. This is because these instruments are located outside the containment and would not be affected by the harsh environment produced inside containment during an accident.

(2) With respect to the availability of instrumentation referred to in the EOP, one or more of the following conditions were met in all cases:

   (a) The preferred indication is qualified and is not routed through the affected Bunker Ramo assembly, and is not affected by the potential qualification deficiency;

   (b) The designated, qualified backup instrument is available to provide the information; and

   (c) The procedures provide alternative actions in the event that the item of information cannot be obtained. These actions are conservative with respect to maintaining critical safety functions (e.g., maintaining ECCS flow if termination conditions cannot be satisfied).

(3) In simulator exercises, operators were able to successfully recover from simulated accidents and establish a controlled cooldown that would eventually lead to initiation of cooling by the residual heat removal system.

The staff divided its review of the licensee's request into two phases. The first was review of available information related to the aspects of environmental qualification of the Bunker Ramo containment penetration assemblies. The second was a review of automatic and manual actions necessary to bring the plant to a safe shutdown condition assuming the loss or failure of all circuits routed through the Bunker Ramo penetrations. The staff reviewed additional information, not available during the inspection and determined that environmental qualification of the Bunker Ramo assemblies had not been demonstrated. This review was focused on two major test programs, the Midland tests and the Braidwood tests. The test data was reviewed from the viewpoint of whether the penetration assemblies would perform their intended function during and after experiencing the harsh environment of a LOCA event. The test results do not establish acceptable IR readings since (1) one of the tests had numerous failures but included terminal blocks and (2) the Midland test failed to demonstrate acceptable IR during the test due to a lack of IR measurements (the time dependence of IR was not established).

The staff views the Midland tests as a basis for evaluating operability of Braidwood Unit 2 configuration after 16 hours into a LOCA events. This restriction is due to the lack of performance information (i.e., IR measurements) during the first 16 hours of the simulated event. After this time, all measurements were found to be acceptable (i.e., 1.0 megaohm).

The licensee also asserted that information available for each component in the assemblies when combined with the assembly test results demonstrated the qualification of the Braidwood Unit 2 configuration. The staff considers this approach to be unacceptable for vital electrical equipment qualification such as containment penetration assemblies. However, this information is applicable when evaluating the operability of the Braidwood Unit 2 Bunker Ramo assemblies. The information concerning each component indicates that the splices and connecting wires have been environmentally qualified and penetration modules similar to the Bunker Ramo modules have been environmentally qualified. Therefore, the staff concludes that the Braidwood Unit 2 penetration assemblies are likely to be operable when exposed to a harsh environment.

The licensee has attributed the low IR measurement obtained in the test of the Bunker Ramo assembly to IR degradation due to terminal blocks used in the test. The staff has reviewed available research information and the results of tests of terminal block performance in a simulated LOCA environment performed by Sandia National Laboratory (NUREG/CR-1692 and NUREG/CR-3691). The Sandia tests covered several makes and models of terminal blocks used in operating nuclear power plants. The results show a significant amount of data in which the terminal block IR degrades to values in the range of 1 to 10 kohm. The staff also concludes that the terminal blocks were the likely source of low IR observed in the Bunker Ramo test.

Based on the above, the staff concludes that the penetration assemblies as installed at Braidwood Unit 2 have a reasonable probability of functioning when exposed to a harsh environment. However, due to the presence of three tests which showed early failure, the staff cannot entirely dismiss the possibility of IR values going below the threshold value of 1.0 megaohm. As a result, the staff evaluated the ability of the plant to safely go to safe shutdown without use of the instrumentation going through the effected penetrations.

Automatic Signals for RPS and ESFAS Functions

The staff has reviewed the analyses of those events which may cause a harsh environment in the containment (loss of coolant, steamline break and feedwater line break accidents) to verify that even without the instrumentation circuits routed through the Bunker Ramo penetrations, RPS and ESFAS initiation will occur automatically when required. Based on this review, the staff agrees with the licensee's conclusion that containment pressure and steamline pressure instrument channels routed through qualified penetration and would provide the necessary automatic actuation even if the instrumentation in the Bunker Ramo penetrations failed. Additionally, the staff believes that it is reasonable to assume that the Bunker Ramo penetrations would not degrade instantaneously (i.e within the time the actuations are required); and because of that there is a high likelihood that instrumentation in the Bunker Ramo penetrations would also provide the necessary RPS and ESFAS actuations.

Instrumentation and Emergency Operating Procedures Necessary to Shut the Plant Down Safely

Those instruments which are important for shutting the plant down safely and which may provide erroneous readings due to faulted penetrations include pressurizer pressure, pressurizer level and steam generator level instruments. However, operator actions necessary to bring the plant to a safe shutdown condition can be accomplished without the pressurizer pressure and level instrumentation by utilizing alternate instrumentation not affected by Bunker Ramo penetration failure. This alternate instrumentation includes wide-range pressure, which serves as an alternate for pressurizer pressure; and cold calibration channel pressurizer level augmented with pressurizer steam space, liquid space and surge line temperature, which serves as an alternative to normal pressurizer level. In regard to this alternate instrumentation, the licensee has made the following commitments: (1) to develop and make available an aid to be used by operators for converting cold calibration channel pressurizer level measurements to equivalent measurements for hot conditions; and (2) to ensure that the alternative instruments for monitoring pressurizer level have been calibrated within the past 6 months or will be recalibrated. In light of the above the staff finds the licensee's alternative methods for monitoring pressurizer pressure and level to be acceptable. Steam generator level indication could be affected by the electrical penetration problem. To determine the ramifications of penetration failure, the licensee modelled the erroneous indication of steam generator level, along with other erroneous indications, on the Byron/Braidwood training simulator. In exercises with the reprogrammed simulator operators were able to deal with these erroneous indications.

Accident scenarios covered in the exercises included small break LOCA, steamline break and feedwater line break. Steam generator level indication was simulated to read erroneously high. During the exercises operators throttled back auxiliary feedwater flow and maintained approximately 50 gpm per steam generator. Auxiliary feedwater flow was not completely terminated by the operators. By observing steam generator pressure and auxiliary feedwater flow together, operators maintained a steam generator level that led them to avoid the need to enter the emergency procedure for loss of heat sink. In addition, operators were able to successfully recover from the simulated accidents and establish a controlled cooldown rate that would lead to residual heat removal (RHR) cooling.

As a result of the exercises discussed above, the licensee has confirmed that the existing emergency operating procedures are sufficient to guide operators through the appropriate recovery actions. However, in the process, additional insights and alternate indications were identified which will form the basis for a special contingency action procedure the licensee has committed to write and put in place prior to operating the plant above 5% power. This will be a general procedure to address multiple instrument failures and will guide operators to use alternate instruments when necessary.

Based on the actions taken and information provided by the licensee, the staff concludes that together the existing operating procedures, alternate instrumentation and special contingency action procedure for multiple instrument failures provide sufficient guidance to operators for implementation of recovery actions necessary to bring the plant to safe shutdown following a spectrum of primary side or secondary side pipe break accidents.

The Commission's regulations in 10 CFR 50.12, Special Exemptions, state that the Commission may, upon application, grant exemptions from the requirements of the regulations of this part, which are: (1) authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security, and (2) the Commission will not consider granting an exemption unless special circumstances are present. One of the six special circumstances which applies to this exemption is that the exemption would provide only temporary relief from the applicable regulation and the licensee or applicant has made good-faith efforts to comply with the regulation.

The NRC staff has reviewed the licensee's evaluation of the special circumstances relative to 50.12(a)(2)(v) good faith efforts.

In the evaluation of 50.12(a)(2)(v), the exemption would provide temporary relief until no later than startup from a surveillance outage scheduled for January of 1989 - possibly sooner if any unscheduled outage of sufficient duration occurred. The requested exemption is a one-time schedular exemption from demonstrating conclusively that the four Bunker Ramo containment penetrations in question are environmentally qualified in accordance with the Commission's regulations. The licensee has implemented a program for both replacement of the Bunker Ramo penetration assemblies at Braidwood Unit 2 and qualification testing of the same containment electrical penetration assemblies thought to be in accordance with the requirements of 10 CFR 50.49. However, while the staff has concluded that the licensee has not satisfied current licensing criteria regarding qualification with respect to insulation resistance, they have also concluded that the licensee efforts to qualify the penetration assemblies were made in good faith.

Good-faith efforts have been made to comply with the requirements of 10 CFR 50.49 as they apply to the containment penetration assemblies in that: (1) the licensee implemented an environmental qualification program approved by the NRC staff; (2) that program utilized detailed methods and procedures for testing electrical containment penetrations thought to be acceptable under applicable licensing criteria and consistent with industry practice at the time; (3) the licensee provided the NRC staff with information in support of their methods and procedures for qualification when documentation regarding qualification was questioned by the NRC.

The staff has evaluated information presented by the licensee in support of its request for a temporary exemption from the requirements in 10 CFR 50.49(f) and 50.49(j) and other relevant information. The evaluation concludes that (1) test data directly applicable to the Braidwood plant is insufficient to support a finding that the Bunker Ramo penetration assemblies have been qualified; (2) there is reasonable assurance that the Bunker Ramo penetration assemblies in Braidwood Unit 2 will function in an acceptable manner following an accident which creates a harsh environment inside containment; (3) alternate instrumentation will be available to serve RPS and ESFAS functions and post-accident monitoring should instruments connected through the Bunker Ramo penetration assemblies become degraded or fail; (4) emergency operating procedures currently in place and newly developed contingency procedures can be successfully implemented in an emergency to shut the plant down safely without instrumentation connected through the Bunker Ramo Penetration assemblies should it be necessary. Based on these considerations the staff concludes that operation of Braidwood Unit 2 under the temporary exemption requested by the licensee does not present an undue risk to the public health and safety.

The NRC staff reviewed the licensee's description of the special circumstances relative to this exemption request and determined that special circumstances do exist as required by 10 CFR 50.12. The staff also concludes that operation of Braidwood Unit 2 under the temporary exemption is authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security.

5 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

For nuclear power facilities whose construction permits were issued on or after July 1, 1974, 10 CFR 50.55a paragraph (g)(3) specifies that components shall meet the preservice examination requirements set forth in editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code applied to the construction of the particular component. However, 10 CFR 50.55a paragraph (a)(3) permits alternative requirements to paragraph (g)(3) when authorized by the Director of the Office of Nuclear Reactor Regulation. Paragraph (a)(3) requires that the applicant demonstrate that (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements of this section would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

By letter dated November 20, 1987, Commonwealth Edison Company (the licensee) transmitted to the NRC information related to an ultrasonic (UT) flaw indication that was detected during preservice examination of an elbow-to-loop stop valve weld in the reactor coolant system at Braidwood Station Unit No. 2. The indication detected by UT was compared to shrinkage flaws previously detected by radiography. The licensee concluded that the flaws were the same and were acceptable pursuant to IWB-3112(b) of the 1983 Edition of Section XI of the ASME Boiler and Pressure Vessel Code. IWB-3112(b) states in part that components whose volumetric or surface examination detects indications that meet the nondestructive examination standards of Section III shall be acceptable for Section XI.

The Materials Engineering Branch reviewed the information provided in the November 20, 1987 letter. Additional information regarding the nondestructive examinations performed and description of the flaw indications lead the staff to the conclusions that (1) the indications detected by radiography and UT could be different indications because of their detected orientations, axial and circumferential, respectively, (2) IWB-3112(b) cannot be applied to the existing situation because the nondestructive examinations utilized (radiography and UT) and the acceptance criteria applied to the indications are different, and (3) the indications (UT) reported exceeded the preservice acceptance limits of Section XI of the ASME Code. The licensee was informed by telephone of the staff's conclusions and agreed to perform an evaluation of the flaw in accordance with IWB-3640 in the 1983 Edition, Winter 1983 Addenda of Section XI of the ASME Code.

In a letter dated February 23, 1988, the licensee requested relief from certain PSI requirements for the elbow-to-isolation valve weld. Article IWB-3112 of this Code Edition and Addenda indicates that flaws exceeding the standards of IWB-3500 shall be unacceptable for service unless such flaws are removed or repaired to the extent necessary to meet the flaw indication standards prior to placement of the component in service. In their February 23, 1988 letter, the applicant indicated that the indication in the elbow-to-isolation valve weld exceeded the standards of IWB-3500. In lieu of excavation and weld repair of the flaw, the licensee proposes to leave the flaw in the piping and perform augmented ultrasonic examinations of the flawed area for the next three inspection periods. The proposed alternative is based on an interim fracture mechanics analysis that indicates that the flaw will not grow to an unacceptable size during the life of the plant. The fracture mechanics analysis is documented in Attachment B to the February 23, 1988, letter. Additional information related to the residual stress distribution used in the fatigue flaw growth analysis and the loads used in the fracture mechanics analysis are discussed in a letter from the licensee dated March 4, 1988.

During preservice ultrasonic examination of the reactor coolant system at Braidwood 2, a flaw was detected on the elbow side of the cast austenitic stainless steel elbow-to-loop isolation valve weld. In Attachment B to the licensee's submittal dated February 23, 1988, the flaw size was determined, using ultrasonic examination, to be 1.5 inches long and 0.5 inches deep, oriented circumferentially and very close to the weld root, but not breaking through to the inside surface. Ultrasonic examination to detect the axial component of the flaw was restricted due to the weld crown geometry. However, after reviewing the construction radiographs, it was determined that the flaw extends no greater than 0.8 inches in length axially and 0.51 inches deep.

The austenitic cast stainless steel elbow was fabricated to SA-351 CF8A requirements and the weld was fabricated using a shielded metal arc welding (SMAW) process. To ensure that these materials can maintain the piping system's structural integrity for 40 years of operation, the flaw must be evaluated using the procedures and acceptance criteria in IWB-3640, Section XI, Division 1 of the ASME Code. IWB-3640 requires that SMAW material with flaws must be evaluated to the flaw size limits in the Tables in IWB-3641 and austenitic cast stainless steel material with flaws must have adequate toughness after aging. Based on J-Integral Elastic Plastic Fracture Mechanics (EPFM) methods, aged austenitic cast stainless steel material has adequate toughness when the materials fracture resistance at fracture initiation, J_{IC}, is greater than the calculated J_{app} for the flaw when the loads are applied.

The licensee performed an interim fatigue flaw growth analysis at the nozzle safe-end location rather than at the elbow-to-isolation valve location. The licensee indicates that the final report, which will contain a fatigue flaw growth analysis at the elbow-to-isolation valve location, will be submitted for staff review within 30 days after issuance of the fuel power license. The interim evaluation indicates that the flaw size which contained the flaw after 40 years of operation will meet less than the acceptance criteria in the Tables in IWB-3640 of the ASME Code. The licensee indicates that the loads at nozzle safe-end location exceed those at the elbow-to-isolation valve location. Hence, the flaw evaluation submitted for the safe-end location should conservatively bound the analysis for the flaw in the elbow-to-isolation valve location. Although the amount of flaw growth at elbow-to-isolation valve location may be different than the amount at the nozzle safe-end location, it is not likely that the final flaw size will exceed the limits in IWB-3640, because there is considerable margin between the IWB-3640 allowable flaw size and the final flaw size calculated in this interim report.

The flaw growth calculation included residual weld stresses, but did not include growth from stress corrosion. Growth from stress corrosion need not be considered in evaluation of this flaw because the flaw is located in the primary loop piping, which, in a PWR, is not subject to stress corrosion cracking.

The licensee's analysis indicates that the final flaw size after 40 years of operation is calculated to be less than the allowable limits in IWB-3640 and that for the small flaw size and faulted load conditions, the J_{app} is estimated to be less than the J_{IC} for the aged austenitic cast stainless steels.

The staff performed an EPFM analysis to determine whether after 40 years of operation, the aged cast stainless steel would have sufficient load carrying capability during normal, upset, emergency and faulted conditions. The procedures used in the analysis are documented in NUREG/CR-4572, "NRC Leak-Before-Break (LBB. NRC) Analysis Method for Circumferentially Through-Wall Cracked Pipes Under Axial Plus Bending Loads," May 1986. The fatigue evaluated a through-wall crack with a length of five inches. The staff's assumed size is conservative because the fatigue evaluation indicates that the flaw would only grow to a depth of 33 percent of the wall thickness. The material properties for aged cast stainless steel utilized in the staff's analysis were the J-integral properties used by the staff in its "leak-before-break" analysis of the Braidwood Unit 2 primary loop. The staff's "leak-before-break" analysis of the Braidwood Unit 2 primary loop was contained in a letter to Commonwealth Edison Company dated October 28, 1985. The staff included in its analysis the ASME Code recommended safety factors for normal, upset, emergency and faulted conditions. The staff's analysis indicates that the J_{IC} for the austenitic cast stainless steel which has been aged for 40 years of operation, are greater than the calculated J_{app} during normal, upset, emergency and faulted conditions. Hence, the staff's EPFM analysis indicates that after 40 years of operation the cast stainless steel will have sufficient load carrying capability during normal, upset, emergency and faulted conditions to prevent failure of the flawed pipe. The staff's EPFM analysis indicates that after 40 years of operation the cast stainless steel will have sufficient load carrying capability during normal, upset, emergency and faulted conditions to prevent failure of the flawed pipe.

The staff has concluded the following:

(1) The methods used to calculate flaw growth and to evaluate the flaw size are adequate for determining the effect of the flaw on the structural integrity of the pipe system.

(2) Since the licensee has used bounding loads in this interim analysis and there is considerable margin between the IWB-3640 allowable flaw size and the final calculated flaw size, it is not likely that the flaw growth analysis that is to be completed by the licensee will result in the flaw exceeding the limits of IWB-3640.

(3) The licensee's proposal to perform augmented ultrasonic examination of the flawed area for the next three inspection periods will confirm the results of the fatigue flaw growth analysis.

The licensee's and staff's fracture mechanics analyses demonstrate that for 40 years of operation the flaw will not grow to a size, which will result in the loss of the structural integrity of the elbow-to-isolation weld. The fracture mechanics analyses demonstrate that the flaw need not be removed and that the proposed alternative, augmented ultrasonic examination, provides an acceptable level of quality and safety. The staff concludes that licensee has satisfied the requirements for reactor coolant pressure boundary inservice inspection and testing relative to this issue and therefore, concludes that it is acceptable.

APPENDIX A

CONTINUATION OF CHRONOLOGY OF NRC STAFF RADIOLOGICAL SAFETY REVIEW OF BRAIDWOOD STATION, UNITS 1 AND 2

February 23, 1988	Letter from licensee transmitting relief request 2NR15 and interim report, "Evaluation of Indication in Unit 2 Loop 1 Elbow to Valve Weld Region," for review and acceptance.
February 26, 1988	Letter from licensee transmitting Revision 49 to CE-1-A, "QA Program for Nuclear Generating Program."
March 4, 1988	Letter from licensee providing further basis for fatigue crack growth evaluation supporting February 23, 1988 request for relief from preservice inspection requirements of ASME Section XI.
March 8, 1988	Letter from licensee transmitting Sargent & Lundy Revision 3 to "ATWS Mitigation System Specific Design for Byron/Braidwood Stations."
March 14, 1988	Letter from licensee providing status of actions taken, per licensee commitment letters attached to License NPF-75, including completion of stated retests and Diesel Generator 2A Technical Specifications.
March 15, 1988	Letter from licensee providing onsite property damage insurance at listed facilities, per 10 CFR 50.54(w)(4).
March 16, 1988	Letter from licensee transmitting reponse to Generic Letter 88-02, "ISAP II."
March 17, 1988	Letter to licensee transmitting Generic Letter 88-05 to all power reactor licensees and applicants.
March 18, 1988	Letter to licensee transmitting notice of consideration of issuance of amendments to Licenses NPF-37, NPF-66, NPF-72, and NPF-66 and Opportunity for a Hearing.
March 22, 1988	Letter to licensee transmitting Generic Letter 88-06 to all power reactor licensee and applicants.
March 22, 1988	Letter to licensee transmitting Amendments 15 (Byron Unit 1), 15 (Byron Unit 2), 6 (Braidwood Unit 1) and (Braidwood Unit 2) to Licenses NPF-37, NPF-66, NPF-72 and NPF-75.

March 23, 1988	Letter from licensee transmitting additional information to support environmental qualification per 10 CFR 50.49 of Bunker Ramo.
March 25, 1988	Letter to licensee transmitting notice of issuance of Amendments 15 (Byron, Unit 1), 15 (Byron, Unit 2) to Licenses NPF-37 and NPF-66 and Amendments 6 (Braidwood, Unit 1), to 6 (Braidwood, Unit 2) to Licenses NPF-72 and NPF-75.
March 29, 1988	Letter to licensee asking receipt of September 11, 1987 response to NRC Bulletin 87-001.
March 31, 1988	Letter from licensee advising of completion of structural steel fireproofing in lower cable spreading room.
April 7, 1988	Letter to licensee transmitting Generic Letter 88-07 to power reactor licensees and applicants.
April 7, 1988	Letter from licensee transmitting additional documentation supporting environmental qualification of Bunker Ramo containment penetrations.
April 7, 1988	Letter from licensee requesting schedular exemption from 10 CFR 50.49(j) requirements.
April 8, 1988	Letter to licensee advising that based on review of additional information provided during March 9 and 16, 1988 meetings and March 28, 1988 letter, NRC concludes that environmental qualification (EQ) has still not been demonstrated for assemblies.
April 8, 1988	Letter to licensee transmitting Amendments 16 (Byron, Unit 1) and 16 (Byron, Unit 2) to Licenses NPF-37 and NPF-66 and Amendments 7 (Braidwood, Unit 1) and 7 (Braidwood, Unit 2) to Licenses NPF-72 and NPF-75.
April 11, 1988	Letter from licensee transmitting amendments and supplemental information regarding Limitorque operator lubrication.
April 15, 1988	Letter to licensee transmitting Amendments 17 (Byron, Unit 1), 17 (Byron, Unit 2), 8 (Braidwood, Unit 1) and 8 (Braidwood, Unit 2) to Licenses NPF-37, NPF-66, NPF-72 and NPF-75.
April 15, 1988	Letter from licensee providing additional information to facilitate NRC review of April 7, 1988 request for schedular exemption from requirements of 10 CFR 50.49(j) regarding Bunker Ramo electrical penetration assemblies.
April 18, 1988	Letter from licensee transmitting Revision 50 to CE-1-A, "QA Program for Nuclear Generating Stations."

April 19, 1988	Letter from licensee transmitting safeguards event logs for January-March 1988.
April 21, 1988	Letter from licensee transmitting supplemental information and errata sheet correcting Action Item 4 of April 4, 1988 letter regarding Limitorque operator lubrication.
April 22, 1988	Letter from licensee correcting error in S. C. Hunsader April 22, 1988 letter to T. E. Murley ragarding Mode 2 date of April 23, 1988 for Unit 1.
April 26, 1988	Letter to licensee notifying that submittals regarding allegations concerning use of poor materials and poor manufacturing practices by C & S Valve Company have been reviewed.
April 28, 1988	Letter to licensee requesting fee for March 29, 1988 application for review of program to validate adequacy of Limitorque operation lubrication.

APPENDIX B

Bibliography

1 - EPRI NP-4916, "Lubrication Guide," January 1987

2 - ASTM D 217-82, "Standard Test Method for Core Penetration of Lubricating Grease."

APPENDIX F

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John A. Kudrick	Section Leader	Plant Systems Branch
Linda L. Luther	Licensing Assistant	Project Directorate III-2
Stephen P. Sands	Project Manager	Project Directorate III-2

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U.S. NUCLEAR REGULATORY COMMISSION

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NUREG-1002
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2. TITLE AND SUBTITLE

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13. ABSTRACT (200 words or less)

In November 1983, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-1002) regarding the application filed by the Commonwealth Edison Company, as applicant and owner, for a license to operate Braidwood Station, Units 1 and 2 (Docket Nos. 50-456 and 50-457). The first supplement to NUREG-1002 was issued in September 1986; the second supplement was issued in October 1986; the third supplement was issued in May 1987; the fourth supplement was issued in July 1987 in support of the full-power license for Unit 1; the fifth supplement was issued in December 1987 in support of the low-power license for Unit 2. This sixth supplement to NUREG-1002 is in support of the full-power license for Unit 2 and provides the status of items that remained unresolved at the time Supplement 5 was published. The facility is located in Reed Township, Will County, Illinois.

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS

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15. AVAILABILITY STATEMENT

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