CENTRAL FILES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

Docket No. 50-410

24 NOV 1978

Niagara Mohawk Power Corporation
ATTN: Mr. G. K. Rhode
Vice President
System Project Management
300 Erie Boulevard, West
Syracuse, NY 13202

Gentlemen:

Enclosed is a supplement (IE Bulletin No. 78-12A) to IE Bulletin No. 78-12 which requires action by you with regard to your power reactor facility(ies) with an operating license or a construction permit.

Should you have any questions regarding this Bulletin or the actions required by you, please contact this office.

Sincerely,

James M. Allan

Boyce H. Grier
~~Director~~

Enclosures:
1. IE Bulletin 78-12A
2. List of IE Bulletins
   Issued in 1978

cc w/encls:
Eugene B. Thomas, Jr., Esquire

7812150171

ma15
a02
R6

[illegible]

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

IE Bulletin No. 78-12A
Date: November 24, 1978

ATYPICAL WELD MATERIAL IN REACTOR PRESSURE VESSEL WELDS

Description of Circumstances:

This Bulletin is a supplement to IE Bulletin 78-12, issued on September 29, 1978, and the two documents should be considered together.

Bulletin 78-12 described the use of weld wire that failed to meet all specified chemical properties in welds of twelve identified reactor pressure vessels. Use of the atypical weld material in vessel weldments causes them to have higher than normal nil-ductility transition temperature characteristics which in turn requires more conservative pressure/temperature operating limits.

Bulletin 78-12 was issued for the purpose of verifying that similar atypical weld material was not also supplied to other vessel manufacturers and used in reactor pressure vessel fabrication. Recognizing that the scope of the record review required is extensive and time consuming, and to assure that responses provided are meaningful, the requirements of Bulletin 78-12 are being modified.

Action To Be Taken By Licensees and Permit Holders:

For all power reactor facilities with an operating license or a construction permit, except those already identified as possibly having atypical weld material[1]:

1. Provide all information available on weld materials used for each reactor vessel primary boundary ferritic weldment.[2] (Items 1c, 1d, 2a, 2b, first sentence of 2c, 3 and 4 of Bulletin 78-12.) This information may be provided to NRC through the vessel manufacturers or suppliers as appropriate to prevent duplication of data.

---

1 The twelve nuclear units identified as having possible atypical pressure vessel weldments are: Three Mile Island Unit Nos. 1 and 2, Crystal River Unit No. 3, Arkansas Nuclear One Unit No. 1, Oconee Unit No. 3, Rancho Seco Unit No. 1, Midland Unit No. 1, Quad Cities Unit No. 2, Browns Ferry Unit No. 1, Turkey Point Unit No. 4 and Zion Unit Nos. 1 and 2.

2 Weld material information submitted will be evaluated by NRC. Requests for further information will be dependent upon results of these evaluations. Additional requests or instructions will be issued following these evaluations.

2. Correlation of specific heat, lot or batch to specific weldments in specific vessels is not required at this time. (Last sentence of Item 2c, Bulletin 78-12.) However, each licensee is required to verify that the weld materials information provided to the NRC under Item 1 does in fact cover each reactor vessel for which the licensee is responsible.

3. Responses to item 1 above shall be submitted in writing within 120 days of the date of this Bulletin supplement. Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, Division of Reactor Construction Inspection, Washington, D.C. 20555.

Approved by GAO, B180225 (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

LISTING OF IE BULLETINS
ISSUED IN 1978

Bulletin No.	Subject	Date Issued	Issued To
78-01	Flammable Contact - Arm Retainers in G.E. CR120A Relays	1/16/78	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
78-02	Terminal Block Qualification	1/30/78	All Power Reactor Facilities with an OL or CP
78-03	Potential Explosive Gas Mixture Accumulations Associated with BWR Offgas System Operations	2/8/78	All BWR Power Reactor Facilities with an OL or CP
78-04	Environmental Qualification of Certain Stem Mounted Limit Switches Inside Reactor Containment	2/21/78	All Power Reactor Facilities with an OL or CP
78-05	Malfunctioning of Circuit Breaker Auxiliary Contact Mechanism-General Model CR105X	4/14/78	All Power Reactor Facilities with an OL or CP
78-06	Defective Cutler-Hammer, Type M Relays With DC Coils	5/31/78	All Power Reactor Facilities with an OL or CP
78-07	Protection afforded by Air-Line Respirators and Supplied-Air Hoods	6/12/78	All Power Reactor Facilities with an OL, all Class E and F Research Reactors with an OL, all Fuel Cycle Facilities with an OL, and all Priority 1 Material Licensees

LISTING OF IE BULLETINS
ISSUED IN 1978

Bulletin No.	Subject	Date Issued	Issued To
78-08	Radiation Levels from Fuel Element Transfer Tubes	6/12/78	All Power, Test, and Research Reactor Facilities with an OL having Fuel Element Transfer Tubes.
78-09	BWR Drywell Leakage Paths Associated with Inadequate Drywell Closures	6/14/78	All BWR Power Reactor Facilities with an OL (for action) or CP (for information).
78-10	Bergen-Paterson Hydraulic Shock Suppressor Accumulator Spring Coils	6/27/78	All Power Reactor Facilities with an OL or CP.
78-11	Examination of Mark I Containment Torus Welds	7/21/78	BWR Power Reactor Facilities with an OL for action: Peach Bottom 2 and 3, Quad Cities 1 and 2, Hatch 1, Monticello and Vermont Yankee. All other BWR Power Reactor Facilities with an OL for information.
78-12	Atypical Weld Material in Reactor Pressure Vessel Welds	9/29/78	All Power Reactor Facilities with an OL or CP
78-13	Failures in Source Heads of Kay-Ray, Inc. Gauges Models 7050, 7050B, 7051, 7051B, 7060, 7060B, 7061 and 7061B	10/27/78	All General and Specific Licensees with the Subject Kay-Ray, Inc. Gauges

CENTRAL FILES

NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

January 23, 1978

Office of Inspection and Enforcement
Region I
Attn: Mr. Boyce H. Grier, Director
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Re: Nine Mile Point Unit 2
Docket No. 50-410

Dear Mr. Grier:

Your December 19, 1977 I.E. Bulletin 77-07 described deficiencies in certain General Electric containment electrical penetration assemblies. The following information regarding electrical penetrations used at Nine Mile Point Unit 2 is provided in response to your requests.

Question 1.0

Do you have containment electrical penetrations that are of the G.E. Series 100, or are otherwise similar in that they depend upon an epoxy sealant and a dry nitrogen pressure environment to ensure the functional capability as required by the plant's safety analysis report; namely, (1) to ensure adequate functioning of electrical safety-related equipment and (2) to ensure containment leak tightness? If you do use penetrations of this type at your facility, describe the manufacturer and model number of these units.

Response

Type G.E. series containment electrical penetrations, or similar epoxy sealant-type penetrations will not be used at Nine Mile Point Unit 2. Additionally, the penetrations that will be used do not require nitrogen gas pressure to perform their function. Nitrogen pressurization of the penetrations is used as a medium to monitor containment leakage.

W

Question 3.0

Is there a need, as determined by either the vendor or yourself, to maintain penetrations pressurized during normal operation, to assure functionability during a LOCA.

Response

The penetrations are not required to be pressurized to perform their function during loss of coolant accident conditions.

Question 3.1

What measures have you taken to ensure that penetrations of this type will perform their design function under LOCA conditions? (design reviews, analyses, or tests)?

Response

Prior to their use, assemblies similar to those planned for use will be qualified to ensure they will perform their design function under loss of coolant accident conditions.

Question 3.2

Are the measures that provide this assurance adequate to satisfy the Commission's regulations (GDC 4, Appendix A to Part 50; QA Criteria, Appendix B to Part 50)?

Response

The electrical penetrations will meet the requirements specified in 10CFR50 Appendix A (General Design Criteria 4) and 10CFR50 Appendix B.

Very truly yours,

NIAGARA MOHAWK POWER CORPORATION

Gerald K. Rhode, Vice President
System Project Management

PEF/szd

Question 1.1

If you do not have penetration assemblies of the type(s) referenced in Item 1.0 above, describe the type(s) of penetrations e.g., manufacturer and model number now in use or planned for use in safety systems at your facility.

Response

Conax modular flange header series with bolt-on flange interfaces to the primary containment penetration nozzle will be used in safety systems at Nine Mile Point Unit 2.

Question 1.2

Do the transition connector pins imbedded in the epoxy as discussed in Item 1.0 above, have an insulation jacket?

Response

The Conax electrical penetrations do not contain connector pins embedded in an epoxy sealant.

Question 2.0

For those penetrations referenced in Item 1 above, has the manufacturer's prescribed nitrogen pressure been maintained at all times during shipping, storage, and installation?

Response

The Conax electrical penetrations will be pressurized with dry nitrogen during shipping and storage operations. Additionally, they will be kept pressurized during and after installation. However, the assemblies do not require pressurization to perform their intended functions.

CENTRAL FILES

TO WHOM IT MAY CONCERN:

THE ATTACHED PAGE IS FORWARDED FOR INCORPORATION WITH NIAGARA MOHAWK (50-410 NINE MILE 2) IE BULLETIN 77-07 RESPONSE DATED JANUARY 23, 1978. IT SHOULD BE INCORPORATED INTO THE RESPONSE LETTER AS PAGE 2. THANK YOU.

CENTRAL FILES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

January 16, 1978

Docket No. 50-410

Niagara Mohawk Power Corporation
ATTN: Mr. G. K. Rhode
Vice President
System Project Management
300 Erie Boulevard, West
Syracuse, NY 13202

Gentlemen:

Enclosed is IE Bulletin No. 78-01 which requires action by you with regard to your power reactor facility(ies) having an operating license or a construction permit.

Should you have questions regarding this Bulletin or the actions required, please contact this office.

Sincerely,

Robert T. Carlson
for Boyce H. Grier
Director

Enclosures:
1. IE Bulletin No. 78-01
2. List of IE Bulletins Issued During Last 12 Months

cc w/encls:
Eugene B. Thomas, Jr., Esquire

W

[illegible]

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

IE Bulletin No. 78-01
Date: January 16, 1978

FLAMMABLE CONTACT-ARM RETAINERS IN G.E. CR120A RELAYS

Description of Circumstances:

On April 18, 1977, a fire was discovered in a relay cabinet in the Peach Bottom Unit 3 facility. The fire damaged eight relays, type CR120A, manufactured by the General Electric Company. The fire was investigated by the licensee, the Philadelphia Electric Company, who concluded that it was caused by an overheated relay coil that ignited the relay's plastic contact-arm retainer. Subsequently, on July 29, 1977, fire damage was discovered in a relay cabinet during surveillance testing in the Peach Bottom Unit 2 facility. This fire damaged 18 G.E. type CR120A relays.

Discussion:

The investigation revealed that the relay's contact-arm retainer was made from Celcon M90 Acetal Copolymer which is a flammable material. In June 1972, this material was changed to Valox 310-SED, which is self extinguishing and flame resistant. Both materials are white and have the same surface appearance. A G.E. Service Information Letter SIL-NO-229 (a copy of the text is attached for information) indicates that there is no generic overheating problem with this type relay. However, they recommend that BWR owners replace the contact-arm retainer of all relays (including BOP applications) marked with manufacturer's date code* between E.D. (May 1968) and A.J. (January 1973) with the improved, self extinguishing flame resistant contact arm retainers.

Action To Be Taken By Licensees And Permit Holders:

Licensees of power reactor facilities with an operating license and construction permit holders shall take the following actions:

1. Determine if you have installed G.E. type CR120A relays in safety-related equipment or in areas wherein fires have the potential for damaging safety equipment. Also determine if you have such relays in spares inventory or on order.

* The first letter signifies the month, the second letter signifies the year. E, the fifth letter, corresponds to the fifth month. Yearly code is as follows: D - 1968; E - 1969; F - 1970; G - 1971; H - 1972; and J - 1973.

2. Identify all of the relays that have Celcon contact-arm retainers.

3. For those relays which have Celcon retainers, develop a program for their replacement with Valox retainers. The program should include:

   a. Identification of the location of the relays

   b. The schedule for the replacement of the Celcon retainers

   c. The procedure that will be used to perform the replacement, including the means that you will use to differentiate between the Valox and Celcon retainers.

4. For facilities with an operating license, a report of the above actions, including the date(s) when they will be completed, shall be submitted within 30 days of receipt of this Bulletin.

5. For facilities with a construction permit, a report of the above actions, including the date(s) when they will be completed, shall be submitted within 60 days of receipt of this Bulletin.

Reports should be submitted to the Director of the appropriate NRC Regional Office. A copy of your report should be sent to the U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

Approved by GAO, B-180225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

Attachment: Copy of Text of General
Electric Service Instruction
Letter (SIL)-SIL-No. 229

(COPY OF TEXT OF GE SIL-No. 229)
FWD. BY LTR MFN 373-77
8-10-77

IMPROVED CONTACT ARM RETAINERS FOR TYPE CR120A RELAYS

Recently, at an operating BWR a small electrical fire occurred in a relay panel. The purpose of this Service Information Letter is to discuss this occurrence and to recommend an improved contact arm retainer for certain CR120A relays to help avoid or mitigate this type of occurrence.

DISCUSSION

In a relay panel at an operating BWR, one relay, Type CR120A, overheated, subsequently ignited, and resulted in seven other similar relays in the proximity being burned. No definite cause could be determined for the relay overheating. Moreover, subsequent investigation concluded that there is no generic overheating problem with relays of this type.

The investigation also noted that the contact arm retainers of these relays, and of relays manufactured between May 1968 and June 1972, were made of Celcon M90 acetal copolymer which is flammable. In June 1972 this material was changed to Valoz 310-SEO which is self extinguishing and flame resistant.

RECOMMENDED ACTION

General Electric recommends that BWR owners replace the contact arm retainers of all Type CR120A relays (including those in B.O.P. applications) marked with a manufacturing date code between ED (May 1968) and AJ (January 1973) with the improved, self extinguishing flame resistant contact arm retainers, as follows:

1. For four pole CR120A relays use replacement contact arm retainer Part No. 55650378P3.

2. For two pole adder CR120A relays use replacement contact arm retainer Part No. 55501383P3.

3. For two pole CR120A relays use replacement contact arm retainer Part No. 55651401P3.

Attachment

(COPY)

BWR owners should survey CR120A relay applications throughout their plants, in both the nuclear steam supply systems (NSSS) and balance of-plant (BOP), to ascertain the total amount of contact arm retainers required for each type of relay. Upon notification of their requirements, NEDs' Spare and Renewal Parts service will furnish BWR owners the requested amount of replacement parts.

Contact your local General Electric service representative for additional information and for assistance in obtaining the recommended replacement parts.

W:caj/70

(COPY)

LISTING OF IE BULLETINS ISSUED IN 1977

Bulletin No.	Subject	Date Issued	Issued To
77-08	Assurance of Safety and Safeguards During an Emergency - Locking Systems	12/28/77	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
77-07	Containment Electrical Penetration Assemblies at Nuclear Power Plants Under Construction	12/19/77	All Power Reactor Facilities with a Construction Permit
77-06	Potential Problems with Containment Electrical Penetration Assemblies	11/22/77	All Power Reactor Facilities with an Operating License (OL)
77-05A	Supplement 77-05A to IE Bulletin No. 77-05 - Electrical Connector Assemblies	11/15/77	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
77-05	Electrical Connector Assemblies	11/8/77	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
77-04	Calculational Error Affecting the Design Performance of a System for Controlling pH of Containment Sump Water Following a LOCA	11/4/77	All PWR Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)

LISTING OF IE BULLETINS
ISSUED IN 1977

Bulletin No.	Subject	Date Issued	Issued To
77-03	On-Line Testing of the W Solid State Protection System	9/12/77	All W Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
77-02	Potential Failure Mechanism in Certain W AR Relays with Latch Attachments	9/12/77	All Holders of Operating Licenses (OL) or Construction Permits (CP)
77-01	Pneumatic Time Delay Relay Set Point Drift	4/29/77	All Holders of Operating Licenses (OL) or Construction Permit (CP)

CENTRAL FILES

NIAGARA MOHAWK

Gerald K. Rhode
Vice President
System Project
Management

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

January 5, 1978

Office of Inspection and Enforcement
Region I
Attn: Mr. Boyce H. Grier, Director
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Re: Nine Mile Point Unit 2
Docket No. 50-410

Dear Mr. Grier:

Your IE Bulletins 77-05, dated November 8, 1977, and 77-05A, dated November 15, 1977, described potential deficiencies in certain pin-type electrical connector assemblies. We have reviewed our design and plan to use similar pin-type electrical connectors in safety-related systems for Nine Mile Point Unit 2.

The adequacy of qualification for Class IE connectors will be reviewed to insure that they fulfill the requirements for their service environment.

Very truly yours,

NIAGARA MOHAWK POWER CORPORATION

Gerald K. Rhode

PEF/szd

7801

CENTRAL FILES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

December 28, 1977

Docket No. 50-410

Niagara Mohawk Power Corporation
ATTN: Mr. G. K. Rhode
Vice President
System Project Management
300 Erie Boulevard, West
Syracuse, NY 13202

Gentlemen:

Enclosed is IE Bulletin No. 77-08 which requires action by you with regard to your power reactor facility(ies) with an operating license or a construction permit.

Should you have questions regarding this Bulletin or the actions required of you, please contact this office.

Sincerely,

R. C. Haynes
for Boyce H. Grier
Director

Enclosures:
1. IE Bulletin No. 77-08
2. List of IE Bulletins
Issued in 1977

cc w/encls:
Eugene B. Thomas, Jr., Esquire

[illegible]

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

ASSURANCE OF SAFETY AND SAFEGUARDS DURING AN EMERGENCY - LOCKING SYSTEMS

Description of Circumstances:

Under emergency conditions, prompt ingress into certain safety-related areas must be assured to enable safe shutdown of a nuclear power plant, and unimpeded egress from all parts of the facility must be assured in the interest of life safety. The circumstances described below indicate that prompt ingress and unimpeded egress under emergency conditions may not be assured at all nuclear power plants.

At one nuclear power plant, upon loss of offsite power resulting in a scram of the reactor, all electrically locked doors to vital areas failed for lack of auxiliary power. (Although, the electrical circuit blue prints indicated that the electrical locking system was connected to the vital bus to provide uninterrupted auxiliary power, the control console for the locking system had not in fact been so connected.) This failure delayed ingress by operations personnel into several safety-related areas because they had to await arrival of a guard with the one immediately available key. Other security keys were at the facility but were either secured or held by a person who was unaware of what the keys would unlock.

Concurrent with the above situation, three employees were isolated without an adequate emergency escape route available to them. The two accessible doors on that level had been secured, one by a failed electrical locking device and the other by a lock which could be opened only by the grand-master key which they did not possess. Further, the second door was blocked from the opposite side. The only other escape route which could be considered was an unenclosed stairwell leading to other levels, but it was blocked by hot water flowing from the turbine floor above. The employees telephoned for assistance and were released by a guard who came through the cable spreading room and opened the failed door from within.

During an NRC inspection at another nuclear power plant it was observed that two exterior emergency exit doors were chained and padlocked from within. Although the padlocks were of the "breakable" shackle type, substantial force would be required to break them and unimpeded egress in an emergency was not assured.

At a third nuclear plant, a technician conducting tests accidently caused a scram, turbine trip, and loss of station power. Some electrical locking devices securing safety-related areas were supplied only from non-vital buses which were stripped of their loads in the process of transferring to secondary power sources. The electrical locking devices failed and delayed the ingress of additional plant personnel to assist in the shutdown of the plant.

Finally, information available to the NRC indicates that licensees at many other nuclear power plants utilize or plan to utilize electrical locking devices for vital areas, protected areas, and non-security areas. Some of the plants do not have auxiliary power for a portion of or all of the electrical locking systems, and these systems could fail in such a way that prompt ingress or unimpeded egress would not be assured.

Discussion of Applicable Requirements:

Appendix E of 10 CFR Part 50 provides that (a) the capability for plant evacuation, and (b) the capability for facility reentry in order to mitigate the consequences of an accident or, if appropriate, to continue operations, must be assured.

Electrical locks not provided with auxiliary power cannot be maintained in an operable condition (10 CFR 73.55(g)(1)), and electrical locks which fail in the open mode are not providing the required locking (73.55(d)(7)). It should be noted that the NRC is currently reviewing amended Security Plans submitted in response to the requirements of 10 CFR 73.55. That review will encompass prompt emergency ingress and unimpeded egress through security related doors in conjunction with positive access controls at facilities having an operating license.

The National Fire Protection Association Standard NFPA 101 is a generally accepted national standard known as the "Life Safety Code." NFPA 101 is the basis of certain regulations of the Occupational Safety and Health Administration (29 CFR 1910) and the fire regulations and life safety codes of a significant number of States. This standard addresses in detail the number, locations, widths, and routes to emergency exits. It further details safety requirements for stairwell escape routes, describes route and exit markings, and specifically instructs against the installation of a lock or other fastening on an emergency exit that would prevent escape from the inside of the building.

Action to be Taken by Licensee and Permit Holders:

1. Survey your facility and facility plans to determine whether the following situations exist:

   a. Prompt emergency ingress into electrically locked safety-related areas by essential personnel is assured in any postulated occurrence through the combined use of features (1), (2), and (3) below or the equivalent.

      (1) Provide reliable and uninterruptable auxiliary power to the entire electrical locking system, including its controls; and

      (2) Provide the electrical locking devices, which are required to fail in the secure mode for security purposes, with secure mechanical means and associated procedures to override the devices upon loss of both primary and auxiliary power (e.g., key locks with keys held by appropriate personnel who know when and how to use them); and

      (3) Provide periodic tests of all locking systems and mechanical overrides to confirm their operability and their capability to switch to auxiliary power.

   b. Unimpeded emergency egress is assured from all parts of your facilities, the security hardware and systems are designed and installed so as to not degrade life safety, and such hardware and systems are in conformance with applicable (State/Local) fire regulations and life safety codes.

2. Review existing emergency plans and procedures to assure that prompt emergency ingress and unimpeded emergency egress are fully and effectively addressed for any postulated occurrence.

3. Assure that prompt emergency ingress and unimpeded egress through security doors at facilities with an operating license are thoroughly described in submittals pursuant to 10 CFR 73.55.

4. In the event that surveys or reviews required by action items 1 and 2 establish that the facility does not meet the requirements noted in these items, holders of an operating license shall provide a written report to the appropriate NRC Regional Office within 45 calendar days of receipt of this Bulletin. The required report will clearly describe all identified problem areas together with proposed corrective actions. Holders of construction permits will respond in like manner within 60 calendar days of receipt of this Bulletin. If your facility is in full conformance with the requirements noted, no response to this Bulletin is required.

Approved by GAO, B180225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

IE Bulletin No. 77-08
December 28, 1977

LISTING OF IE BULLETINS
ISSUED IN 1977

Bulletin No.	Subject	Date Issued	Issued To
77-07	Containment Electrical Penetration Assemblies at Nuclear Power Plants Under Construction	12/19/77	All Power Reactor Facilities with a Construction Permit
77-06	Potential Problems with Containment Electrical Penetration Assemblies	11/22/77	All Power Reactor Facilities with an Operating License (OL)
77-05A	Supplement 77-05A to IE Bulletin No. 77-05 - Electrical Connector Assemblies	11/15/77	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
77-05	Electrical Connector Assemblies	11/8/77	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
77-04	Calculational Error Affecting the Design Performance of a System for Controlling pH of Containment Sump Water Following a LOCA	11/4/77	All PWR Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)

IE Bulletin No. 77-08
December 28, 1977

LISTING OF IE BULLETINS ISSUED IN 1977

Bulletin No.	Subject	Date Issued	Issued To
77-03	On-Line Testing of the W Solid State Protection System	9/12/77	All W Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
77-02	Potential Failure Mechanism in Certain W AR Relays with Latch Attachments	9/12/77	All Holders of Operating Licenses (OL) or Construction Permits (CP)
77-01	Pneumatic Time Delay Relay Set Point Drift	4/29/77	All Holders of Operating Licenses (OL) or Construction Permit (CP)

Enclosure 2

[illegible]

CENTRAL FILES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

December 19, 1977

Docket No. 50-410

Niagara Mohawk Power Corporation
ATTN: Mr. G. K. Rhode
Vice President
System Project Management
300 Erie Boulevard, West
Syracuse, NY 13202

Gentlemen:

Enclosed is IE Bulletin No. 77-07 which requires action by you with regard to your power reactor facility(ies) with a construction permit.

Should you have questions regarding this Bulletin or the actions required of you, please contact this office.

Sincerely,

Robert T. Carlson
for Boyce H. Grier
Director

Enclosures:
1. IE Bulletin 77-07
2. List of IE Bulletins Issued in 1977

cc w/encls:
Eugene B. Thomas, Jr., Esquire

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

CONTAINMENT ELECTRICAL PENETRATION ASSEMBLIES AT NUCLEAR POWER PLANTS UNDER CONSTRUCTION

Description of Circumstances:

On October 3, 1977, Northeast Nuclear Energy Company reported to the NRC Region I Office that two control valves installed inside containment at Millstone Unit No. 2 demonstrated abnormal operational characteristics. The licensee reported that an unexpected closure of a letdown flow stop valve occurred. While investigating this problem, the normally closed safety injection recirculation return line drain valve was found to be in the open position. Investigation of these events revealed the cause for failure to be electrical shorts between conductors within a containment low voltage penetration assembly.

The licensee subsequently determined that the wiring for both of the valves shared the same low voltage module in an electrical penetration. Electrical tests by the licensee revealed that 15 of the 85 conductors in the suspect connector module exhibited decreased insulation resistance between conductors. Based on this finding, it is believed that an electrical path between adjacent circuits in the connector module was established. This resulted in spurious operation of the valves. Similar resistance checks performed on the remaining low voltage modules within the affected penetration assembly revealed 17 additional conductors with reduced insulation resistances. All conductors with resistances less than 20 megohms were disconnected and their circuits were reconnected through spare conductors.

Examination of the three remaining low voltage penetration assemblies identified 7 additional conductors with resistances of less than 20 megohms. Each of these circuits were also reconnected through a spare conductor.

Investigation showed that the reduced insulation resistance was probably caused by moisture accumulation within the penetration assembly together with small fissures in the epoxy seals surrounding each conductor in the module. The licensee believes that moisture penetrating these cracks

reduced the insulation resistance between adjacent conductors. To prevent further degradation from moisture buildup within the penetration assemblies, the licensee re-established a dry nitrogen pressure of 24 PSIG in the penetrations.

Subsequently the licensee reported that a second event of a similar nature occurred on October 14, 1977. In this instance the sample isolation valve for the pressurizer surge line failed to close on command. Investigation into this event indicated that electrical shorts between conductors due to a moisture accumulation problem were the probable cause for valve misoperation. The shorted wires were disconnected and the valve was de-energized in the closed position.

In discussions on the issue with the licensee and the electrical penetration vendor, General Electric Company, NRC staff determined that maintenance of nitrogen pressure is essential to the integrity of both high and low voltage penetration assemblies. The General Electric Company specifies in its penetration assembly maintenance and operation manual that a 15 PSIG dry nitrogen pressure should be maintained on low voltage units while 30 PSIG should be maintained on high voltage units.

Action To Be Taken By Applicants Of all Power Reactor Facilities With a Construction Permit:

Containment Electrical Penetrations - For safety related systems

1.0 Do you have containment electrical penetrations that are of the G. E. Series 100, or are otherwise similar in that they depend upon an epoxy sealant and a dry nitrogen pressure environment to ensure that the electrical and pressure characteristics are maintained so as to ensure the functional capability as required by the plant's safety analysis report; namely, (1) to ensure adequate functioning of electrical safety-related equipment and (2) to ensure containment leak tightness? If you do use penetrations of this type at your facility describe the manufacturer and model number of these units.

1.1 If you do not have penetration assemblies of the type(s) referenced in Item 1.0 above, describe the type(s) of penetrations e.g., manufacturer and model number now in use or planned for use in safety systems at your facility.

1.2 Do the transition connector pins imbedded in the epoxy as discussed in Item 1.0 above, have an insulation jacket?

2.0 For those penetrations referenced in Item 1 above, has the manufacturer's prescribed nitrogen pressure been maintained at all times during shipping, storage, and installation?

3.0 Is there a need, as determined by either the vendor or yourself, to maintain penetrations pressurized during normal operation, to assure functionability during a LOCA.

3.1 What measures have you taken to ensure that penetrations of this type will perform their design function under LOCA conditions? (design reviews, analyses, or tests)?

3.2 Are the measures that provide this assurance adequate to satisfy the Commission's regulations (GDC 4, Appendix A to Part 50; QA Criteria, Appendix B to Part 50)?

4.0 Provide your response to Items 1.0 through 3.2 above in writing within 30 days. Responses should be submitted to the Director of the appropriate NRC Regional Office. A copy of your response should be forwarded to the U. S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, Division of Reactor Construction Inspection, Washington, D. C. 20555.

Approved by GAO, B180225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

IE Bulletin No. 77-07
December 19, 1977

LISTING OF IE BULLETINS
ISSUED IN 1977

Bulletin No.	Subject	Date Issued	Issued To
77-06	Potential Problems with Containment Electrical Penetration Assemblies	11/22/77	All Power Réactor Facilities with an Operating License (OL)
77-05A	Supplement 77-05A to IE Bulletin No. 77-05 - Electrical Connector Assemblies	11/15/77	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
77-05	Electrical Connector Assemblies	11/8/77	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
77-04	Calculational Error Affecting the Design Performance of a System for Controlling pH of Containment Sump Water Following a LOCA	11/4/77	All PWR Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
77-03	On-Line Testing of the W Solid State Protection System	9/12/77	All W Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
77-02	Potential Failure Mechanism in Certain W AR Relays with Latch Attachments	9/12/77	All Holders of Operating Licenses (OL) or Construction Permits (CP)
77-01	Pneumatic Time Delay Relay Set Point Drift	4/29/77	All Holders of Operating Licenses (OL) or Construction Permits (CP)

[illegible]

CENTRAL FILES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

December 13, 1977

Docket No. 50-410

Niagara Mohawk Power Corporation
ATTN: Mr. G. K. Rhode
Vice President
System Project Management
300 Erie Boulevard, West
Syracuse, NY 13202

Gentlemen:

The enclosed IE Circular 77-16 is forwarded to you for information. No written response is required. Should you have any questions related to your understanding of this matter, please contact this office.

Sincerely,

Robert T. Carlson
for Boyce H. Grier
Director

Enclosures:
1. IE Circular 77-16
2. List of IE Circulars Issued in 1977

cc w/encls:
Eugene B. Thomas, Jr., Esquire

A0 2

GD

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

IE Circular 77-16
Date: December 13, 1977

EMERGENCY DIESEL GENERATOR ELECTRICAL TRIP LOCK-OUT FEATURES

Description of Circumstances:

On June 15, 1977, Duquesne Light Company (Beaver Valley 1) reported that during the performance of a test of the diesel-generator (D/G) trip lock-out features in the emergency mode of operation, the D/G output circuit breaker opened when the field voltage trip interlock was tested. This is contrary to a requirement for this facility that, in the emergency mode, all D/G output breaker trips except generator differential and overcurrent be automatically disabled. The engine overspeed trip, which shuts down the diesel engine (but does not affect breaker operation) is also expected to be operable during the emergency mode of operation.

An investigation conducted by the licensee disclosed that the unexpected opening of the output breaker was due to deenergizing a field voltage sensing relay which was supplied by the vendor but had not been disconnected during the on-site acceptance testing of the D/G nor disabled by the protection circuitry logic. A redundant field voltage relay which was supplied by the licensee is correctly by-passed during fast start conditions and emergency operation.

A design change was initiated by the licensee which removed the field voltage trip feature. This was accomplished by disconnecting the set of relay contacts to the trip circuitry of the D/G output breaker. Subsequent testing of the D/G was performed by the licensee which demonstrated satisfactory operation.

This is an example of an event which resulted from inadequate test procedure performance. The procedures as performed had not previously identified the type of deficiency described in this Circular.

The safety significance of this situation is that the premodified protection circuitry would have opened the circuit breaker if a loss of field voltage occurred while running in the emergency mode of operation.

The D/G Units for the above facility were supplied by the Electro Motive Division (EMD) of General Motors. The model numbers for the D/G Units are:

Engine Model No. 20-645-E4
Generator Model No. A-20-C2
Control Panel Model No. 999-20

All holders of operating licenses or construction permits should assure that the appropriate D/G protection trip circuits are provided with automatic by-pass features that prevent them from negating automatic starting or tripping of D/Gs during fast start or emergency operations. It is recommended that the following be considered in your reviews of this matter:

1. Facility procedures should specifically determine whether the protection circuitry that trips the D/G set or the associated output breaker is in accordance with the facility Technical Specifications.

2. Test procedures for your D/G sets (e.g. acceptance preoperational and surveillance tests) should be reviewed to assure that D/G system performance is demonstrated by these tests to be in accordance with related operational requirements specified in the facility Technical Specifications.

3. Strengthening of management controls should be reviewed as necessary to assure adherence to D/G test procedures by plant personnel.

No written response to this Circular is required. If you require additional information regarding this matter, contact the Director of the appropriate NRC Regional Office.

LISTING OF IE CIRCULARS ISSUED IN 1977

CIRCULAR NO.	SUBJECT	DATE OF ISSUE	ISSUED TO
77-01	Malfunctions of Limitorque Valve Operators	1-6-77	All holders of Operating License (OL) or Construction Permit(CP)
77-02	Potential Heavy Spring Flooding	2-18-77	All affected holders of OLs
77-02A	Potential Heavy Spring Flooding	2-18-77	All affected holders of CPs
77-03	Fire Inside a Motor Control Center	3-4-77	All holders of OLs and CPs
77-04	Inadequate Lock Assemblies	3-18-77	Safeguard Group I, II, IV, V, Licensees
77-05	Liquid Entrapment in Valve Bonnets	3-29-77	All holders of OLs and CPs
77-06	Effects of Hydraulic Fluid on Electrical Cable	4-5-77	All holders of OLs and CPs
77-07	Short Period During Reactor Startup	4-14-77	Holders of BWR OLs
77-08	Failure of Feedwater Sample Probe	4-18-77	All holders of OLs
77-09	Improper Fuse Coordination In BWR Standby Liquid Control System Control Circuits	5-27-77	All holders of BWR OLs or CPs
77-10	Vacuum Conditions Resulting in Damage to Liquid Process Tanks	7-15-77	All holders of OLs

LISTING OF IE CIRCULARS ISSUED IN 1977 (Continued)

CIRCULAR NO.	SUBJECT	FIRST DATE OF ISSUE	ISSUED TO
77-11	Leakage of Containment Isolation Valves with Resilient Seats	9-6-77	All holders of OLs and CPs
77-12	Dropped Fuel Assemblies at BWR Facilities	9-20-77	All holders of BWR OLs or CPs
77-13	Reactor Safety Signals Negated During Testing	9-23-77	All holders of OLs and CPs
77-14	Separation of Contaminated Water Systems From Noncontaminated Plant Systems	11-28-77	All Power and Test Reactor, Fuel Cycle, and major By-product material processor facilities with OLs or CPs
77-15	Degradation of Fuel Oil Flow to the Emergency Diesel Generator	11-30-77	All holders of OLs and CPs

[illegible]

CENTRAL FILES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

November 30, 1977

Docket No. 50-410

Niagara Mohawk Power Corporation
ATTN: Mr. G. K. Rhode
Vice President
System Project Management
300 Erie Boulevard, West
Syracuse, NY 13202

Gentlemen:

The enclosed IE Circular 77-15 is forwarded to you for information. No written response is required. Should you have any questions related to your understanding of this matter, please contact this office.

Sincerely,

Robert T. Carlson
for Boyce H. Grier
Director

Enclosures:
1. IE Circular 77-15
2. List of IE Circulars Issued in 1977

cc w/encls:
Eugene B. Thomas, Jr., Esquire

AG 2 of 2
MISC 1
W

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

IE Circular No. 77-15
Date: November 30, 1977

DEGRADATION OF FUEL OIL FLOW TO THE EMERGENCY DIESEL GENERATOR

During surveillance testing on July 14, 1977, personnel at the Cooper Nuclear Station noted a degradation of fuel oil flow to the day tank for the emergency diesel generator. Although the fuel oil transfer pump capacity is 13.8 gpm for each of the two redundant pumps, flow to the day tank for number one diesel generator was only 3 gpm. At full load, engine consumption is 4.5 gpm.

Investigation of this occurrence revealed a clogged strainer in a float operated shutoff valve on the day tank inlet. This valve operates as a backup to level switches which start and stop the fuel oil transfer pumps to maintain normal day tank level. The strainer is an integral part of the float valve assembly and is not shown on the as-built system drawings. This valve was manufactured by McDonnell-Millen Company. Station personnel were thus unaware of the presence of this strainer and did not schedule it for routine strainer cleaning under the preventive maintenance program. Normal testing of the system under the Technical Specification surveillance requirements does not verify system flow rates.

This occurrence represents an example where the as-built system configuration was not accurately indicated on the system drawings, and that adequate system description was apparently not available to Station personnel.

All holders of construction permits or operating licenses should be aware of the potential for variance between as-built configurations and system drawings. This is especially true for support systems to the engineered safeguards features where all required system conditions such as pressure and flows may not receive routine testing under the surveillance testing program. It is recommended that the following be considered in your review of this matter:

1. A field verification of the drawing against the as-built system configuration should be made for the entire diesel generator fuel oil delivery system from the storage tanks to the engines. Appropriate changes should be made to the drawings and preventive maintenance program to account for any components or configurations not previously covered.

2. Consideration should be given to revising surveillance test procedures to include a flow test on the fuel oil system to ensure the system continues to meet design specifications.

In addition, the following information relating to the maintenance of fuel oil cleanliness should be considered in your review:

1. During long-time storage, degradation of fuel oil is a common occurrence. The rate of degradation is not easily predicted since it is a function of the source of the crude oil, the process utilized in making the fuel (e.g., straight distillation or the method of catalytic cracking), and the conditions under which the fuel oil is stored.

2. It is known that certain detrimental processes are accelerated in fuel oils when they are in contact with certain metals. The presence of zinc, such as from galvanizing, has a tendency to form soluble soaps in the fuel oil which are deposited on the diesel engine's injection nozzles. A buildup of this deposit will eventually degrade the engine's performance. The presence of copper promotes the formation of gums which degrade the stored fuel oil and tends to clog filters.

3. The presence of water in the fuel oil promotes the growth of fungi or slime that also degrades the fuel and has the potential for clogging filters.

No written response to this Circular is required. If you require additional information regarding this matter, contact the Director of the appropriate NRC Regional Office.

IE Circular No. 77-15

November 30, 1977

LISTING OF IE CIRCULARS ISSUED IN 1977

CIRCULAR NO.	SUBJECT	DATE OF ISSUE	ISSUED TO
77-01	Malfunctions of Limitorque Valve Operators	1-6-77	All holders of Operating License (OL) or Construction Permit(CP)
77-02	Potential Heavy Spring Flooding	2-18-77	All affected holders of OLs
77-02A	Potential Heavy Spring Flooding	2-18-77	All affected holders of CPs
77-03	Fire Inside a Motor Control Center	3-4-77	All holders of OLs and CPs
77-04	Inadequate Lock Assemblies	3-18-77	Safeguard Group I, II, IV, V, Licensees
77-05	Liquid Entrapment in Valve Bonnets	3-29-77	All holders of OLs and CPs
77-06	Effects of Hydraulic Fluid on Electrical Cable	4-5-77	All holders of OLs and CPs
77-07	Short Period During Reactor Startup	4-14-77	Holders of BWR OLs
77-08	Failure of Feedwater Sample Probe	4-18-77	All holders of OLs
77-09	Improper Fuse Coordination In BWR Standby Liquid Control System Control Circuits	5-27-77	All holders of BWR OLs or CPs
77-10	Vacuum Conditions Resulting in Damage to Liquid Process Tanks	7-15-77	All holders of OLs

LISTING OF IE CIRCULARS ISSUED IN 1977 (Continued)

CIRCULAR NO.	SUBJECT	FIRST DATE OF ISSUE	ISSUED TO
77-11	Leakage of Containment Isolation Valves with Resilient Seats	9-6-77	All holders of OLs and CPs
77-12	Dropped Fuel Assemblies at BWR Facilities	9-20-77	All holders of BWR OLs or CPs
77-13	Reactor Safety Signals Negated During Testing	9-23-77	All holders of OLs and CPs
77-14	Separation of Contaminated Water Systems From Noncontaminated Plant Systems	11-28-77	All Power and Test Reactor, Fuel Cycle, and major By-product material processor facilities with OLs or CPs

Central Files

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

November 28, 1977

Docket No. 50-410

Niagara Mohawk Power Corporation
ATTN: Mr. G. K. Rhode
Vice President
System Project Manager
300 Erie Boulevard, West
Syracuse, NY 13202

Gentlemen:

The enclosed IE Circular 77-14 is forwarded to you for information. No written response is required. Should you have any questions related to your understanding of this matter, please contact this office.

Sincerely,

Robert T. Carlson
for Boyce H. Grier
Director

Enclosures:
1. IE Circular 77-14
2. List of IE Circulars Issued in 1977

cc w/encls:
Eugene B. Thomas, Jr., Esquire

A02OR
MISC1 w/

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

November 28, 1977

IE Circular No. 77-14

SEPARATION OF CONTAMINATED WATER SYSTEMS FROM NONCONTAMINATED PLANT SYSTEMS

This circular describes an event which occurred at a nuclear power facility; however, the generic implications may be applicable to test reactors, fuel cycle facilities, and major by-product material processors.

In June, 1977, the licensee for Beaver Valley 1 reported that make up water from a primary water storage tank (PWST) contaminated the plant water treatment system, which in turn supplies the inplant domestic (potable) water system. The plant domestic water became contaminated with a tritium concentration of 7xE(-3) uCi/ml. The domestic water was contaminated for approximately six hours before the condition was detected by the licensee and controls were established over the use of in-house water. No significant exposure of plant personnel resulted from the event; however, five individuals showed positive levels of tritium by urinalysis. No release to the offsite environment above maximum permissible occurred.

The PWST receives processed reactor coolant water from the Boron Recovery System which has been purified through evaporation, degasification and demineralization to remove radioisotopes other than tritium. The PWST is used to supply primary grade water to the reactor coolant system and is normally kept separated from the water treatment system. The cross connection between the primary grade water system and the water treatment system occurred when an isolation valve was inadvertently left open during valve line-up operations to recirculate the PWST. The procedure which specified the required line-up was being used for the first time since preoperational testing and did not list the subject valve.

In addition to the valving error, however, a design error resulted in connecting a line from the PWST to a water treatment system line at a position upstream of two series stop-check valves. In the proper configuration, the line would have been connected downstream of the stop-check valves, which would have prevented back flow of water from the primary grade water system to the water treatment system even with the isolation valve left open. Corrective actions taken

by the licensee were to: (1) correct identified procedural deficiencies which led to the valving error; (2) modify the piping installation to the intended configuration; and (3) add two series isolation valves upstream of the stop-check valves in a "tell-tale" arrangement to provide an air break between the primary water and water treatment lines. The above corrective actions are being reviewed for suitability with design separation criteria.

Section 10.5.3 of the National Standard Plumbing Code requires double check valves or siphon breaker between potable and nonpotable systems. Section 9.2.4 of the Standard Review Plan (NUREG-75/087) states that the acceptance criteria for design of the potable and sanitary water systems (PSWS) is acceptable if there are no interconnections between the PSWS and systems having the potential for containing radioactive materials.

A somewhat similar incident had previously occurred in March, 1975, at Millstone Units 1 and 2, when an improperly wired conductivity cell instrument permitted the return of high activity water to the house heating boiler makeup system. Overflow from the deaerating feed tank and surge tank, which are components of the house heating boiler makeup system, resulted in an unfiltered and unmonitored release of contaminated water.

It is recommended that you review your systems and as-built (or design) drawings, identify all interconnections between contaminated and non-contaminated water systems, and review the interconnection design to assure that separation has been provided. Operating procedures which could lead to inadvertent contamination of domestic water systems should be reviewed to verify that proper valve lineup and administrative controls are provided to prevent contamination of the domestic water supply and the subsequent intake of radioisotopes by plant personnel.

No written response to this Circular is required. Your review of this matter to determine its applicability to your facility and any corrective and preventive actions taken or planned, as appropriate, will be reviewed during a subsequent NRC inspection. If you desire additional information regarding this matter, contact the Director of the appropriate NRC Regional Office.

IE Circular No. 77-14 . November 28, 1977

LISTING OF IE CIRCULARS ISSUED IN 1977

CIRCULAR NO.	SUBJECT	DATE OF ISSUE	ISSUED TO
77-01	Malfunctions of Limitorque Valve Operators	1-6-77	All holders of Operating License (OL) or Construction Permit(CP)
77-02	Potential Heavy Spring Flooding	2-18-77	All affected holders of OLs
77-02A	Potential Heavy Spring Flooding	2-18-77	All affected holders of CPs
77-03	Fire Inside a Motor Control Center	3-4-77	All holders of OLs and CPs
77-04	Inadequate Lock Assemblies	3-18-77	Safeguard Group I, II, IV, V, Licensees
77-05	Liquid Entrapment in Valve Bonnets	3-29-77	All holders of OLs and CPs
77-06	Effects of Hydraulic Fluid on Electrical Cable	4-5-77	All holders of OLs and CPs
77-07	Short Period During Reactor Startup	4-14-77	Holders of BWR OLs
77-08	Failure of Feedwater Sample Probe	4-18-77	All holders of OLs
77-09	Improper Fuse Coordination In BWR Standby Liquid Control System Control Circuits	5-27-77	All holders of BWR OLs or CPs
77-10	Vacuum Conditions Resulting in Damage to	7-15-77	All holders of OLs

[illegible]

LISTING OF IE CIRCULARS ISSUED IN 1977 (Continued)

CIRCULAR NO.	SUBJECT	FIRST DATE OF ISSUE	ISSUED TO
77-11	Leakage of Containment Isolation Valves with Resilient Seats	9-6-77	All holders of OLs and CPs
77-12	Dropped Fuel Assemblies at BWR Facilities	9-20-77	All holders of BWR OLs or CPs
77-13	Reactor Safety Signals Negated During Testing	9-23-77	All holders of OLs and CPs

[illegible]

CF

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

November 15, 1977

Docket No. 50-410

Niagara Mohawk Power Corporation
ATTN: Mr. G. K. Rhode
Vice President
System Project Manager
300 Erie Boulevard, West
Syracuse, NY 13202

Gentlemen:

Enclosed is a supplement (IE Bulletin No. 77-05A) to IE Bulletin No. 77-05 which requires action by you with regard to your power reactor facility(ies) with an operating license or a construction permit.

Should you have questions regarding this Bulletin or the actions required of you, please contact this office.

Sincerely,

R. C. Haynes
for Boyce H. Grier
Director

Enclosures:
1. IE Bulletin No. 77-05A
2. List of IE Bulletins
   Issued in 1977

cc w/encls:
Eugene B. Thomas, Jr., Esquire

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

November 15, 1977

IE Bulletin 77-05A

ELECTRICAL CONNECTOR ASSEMBLIES

Description of Circumstances:

This Bulletin is a supplement to IE Bulletin 77-05, issued November 8, 1977, and the two documents should be considered together.

Bulletin 77-05 described the failure, under test conditions, of a general type of electrical connector. The tests were intended to simulate post-LOCA conditions, therefore the action requested in the Bulletin focused on the qualification of electrical connectors for use inside containment.

Since issuance of Bulletin 77-05, our attention has been called to circumstances which indicate that the scope of the action requested should be expanded and therefore the response to Bulletin 77-05 should reflect the expanded scope.

Electrical connectors should be qualified to perform their intended function after having been subjected to accident conditions if they are contained in a system whose function is to mitigate that accident.

The location of the connectors that must be qualified is not limited to those inside containment.

Action To Be Taken By Licensees and Permit Holders:

1. Actions requested by Bulletin 77-05 should be expanded to include all connectors in safety systems which are required to function to mitigate an accident where the accident itself could adversely affect the ability of the system to perform its safety function. The examination is not to be limited to only LOCA's nor to areas only within containment.

2. Responses should be provided within 30 or 60 days, as appropriate, of the date of this supplement.

Approved by GAO, B180225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

IE Bulletin 77-05A
November 15, 1977

LISTING OF IE BULLETINS ISSUED IN 1977

Bulletin No.	Subject	Date Issued	Issued To
77-05	Electrical Connector Assemblies	11/8/77	All Holders of Operating Licenses (OL) or Construction Permits (CP)
77-04	Calculational Error Affecting the Design Performance of a System for Controlling pH of Containment Sump Water Following a LOCA	11/4/77	All PWR Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
77-03	On-Line Testing of the W Solid State Protection System	9/12/77	All W Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
77-02	Potential Failure Mechanism in Certain W AR Relays with Latch Attachments	9/12/77	All Holders of Operating Licenses (OL) or Construction Permits (CP)
77-01	Pneumatic Time Delay Relay Set Point Drift	4/29/77	All Holders of Operating Licenses (OL) or Construction Permits (CP)

CENTRAL FILES

NIAGARA MOHAWK

Gerald K. Rhode
Vice President
System Project
Management

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

November 8, 1977

Office of Inspection and Enforcement
Region I
Attn: Mr. Boyce H. Grier, Director
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Re: Nine Mile Point Unit 2
Docket No. 50-410

Dear Mr. Grier:

Your September 12, 1977 IE Bulletin 77-02 described deficiencies in certain Westinghouse "AR" relays. We are pleased to inform you that a review of vendor drawings reveals that these relays are not planned to be used at Nine Mile Point Unit 2.

To ensure that this type of relay will not be used, this relay will be added to the purchase specifications "Excluded Equipment List."

Very truly yours,

NIAGARA MOHAWK POWER CORPORATION

Gerald K. Rhode

NLR/szd

AAR
CF

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

November 8, 1977

Docket No. 50-410

Niagara Mohawk Power Corporation
ATTN: Mr. G. K. Rhode
Vice President
System Project Manager
300 Erie Boulevard, West
Syracuse, NY 13202

Gentlemen:

Enclosed is IE Bulletin No. 77-05 which requires action by you with regard to your power reactor facility(ies) with an operating license or a construction permit.

Should you have questions regarding this Bulletin or the actions required of you, please contact this office.

Sincerely,

Robert T. Carlson
for Boyce H. Grier
Director

Enclosures:
1. IE Bulletin No. 77-05
w/Attachments (2)
2. List of IE Bulletins
Issued in 1977

cc w/encls:
Eugene B. Thomas, Jr., Esquire

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

ELECTRICAL CONNECTOR ASSEMBLIES

Description of Circumstances

Recent tests conducted by the Sandia Laboratories of electrical connector/cable assemblies in a simulated post-LOCA containment environment (LWR) demonstrated that the assemblies failed to perform in an acceptable manner. The connectors are the pin and socket type, with metal shell and screw couplings. The specific test specimens were manufactured by Bendix, ITT Cannon and Gulton Industries using combinations of Anaconda and ITT Surprenant cables. Details of the specific connector/cable combinations, test conditions, test results and other pertinent information are described in the Attachment.

While electrical connectors of the type tested are not normally used in applications that are required to survive LOCA conditions, it is not possible in the absence of specific information to conclude that such applications do not exist. Further, it is unknown whether other manufacturers have supplied similar assemblies, whether such assemblies have been properly qualified for the intended service or whether these type of assemblies are utilized in applications that must continue to operate reliably in a LOCA environment.

Action To Be Taken By Licensees and Permit Holders:

FOR ALL POWER REACTOR FACILITIES WITH AN OPERATING LICENSE OR A CONSTRUCTION PERMIT:

1. Determine whether your facility utilizes or plans to utilize electrical connector assemblies of the type tested by Sandia Laboratories, or any other similar type, in systems that are located inside containment, are subject to a LOCA environment and are required to be operable during a LOCA.

2. If any such applications are identified, review the adequacy of qualification testing for the assemblies and submit the documentation for NRC review.

3. If evidence is not available to support a conclusion of adequacy, submit your plans and programs toward qualifying existing equipment or your plans for replacing unqualified assemblies with qualified equipment.

4. Provide your response in writing within 30 days for facilities with an operating license and within 60 days for facilities with a construction permit. Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the U. S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, Division of Reactor Construction Inspection, Washington, D. C. 20555.

Approved by GAO, B-180225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

Attachments:
A. Trip Report by W. R. Rutherford
   Electrical Connector Assemblies
B. Test Profile, Figures 1 and 2

TRIP REPORT
by
W. R. Rutherford

ELECTRICAL CONNECTOR ASSEMBLIES

On September 1, 1977 a meeting was held in Albuquerque, New Mexico to investigate the electrical connector assembly malfunctions or failures that occurred during tests under LOCA conditions performed by Sandia Laboratories. The following is a description of the equipment, test scope and results of these tests.

Equipment

The test assemblies of particular interest consisted of three types of connectors: Bendix, ITT Cannon, and Gulton installed on two types of cables; Anaconda and ITT Surprenant.

1. Bendix Connector: A 3 conductor/No. 12 AWG with crimp pin conductors, anodized aluminum shell, silicone rubber insert, rigid back plane potting, pliable over-potting.

2. ITT Cannon Connector; A 3 conductor/No. 12 AWG with crimp pin conductors, anodized aluminum shell, silicone rubber insert, anodized aluminum back shell, rubber packing boot, mechanical retaining clamp.

3. Gulton Connector: A 3 conductor/No. 12 AWG with crimp pin conductor, stainless steel shell, hard fiber insert, pin back sealed with RTV 112, stainless shell, back plane poured with Sylgard potting, mechanical clamp termination.

4. Anaconda Cable: A 3 conductor/No. 12 AWG, tinned copper conductor, 30 mil ethylene propylene rubber insulation 15 mil Hypalon jacket, cable asbestos tape; 60 mil Hypalon jacket, rated 600 volts, cable diameter 0.55".

5. ITT Surprenant Cable: A 3 conductor/No. 12 AWG, tinned copper conductor, 30 mil Exane II insulation, silicone glass tape, 65 mil Exane jacket, rated 600 volts, cable diameter 0.455".

Test Scope

The three tests performed by Sandia were composed of two sequential and one simultaneous exposure to LOCA environments. In each case the equipment was exposed to radiation and thermal aging prior to operating under the simulated LOCA conditions. Figures 1 and 2 describe the test profiles for sequential and simultaneous tests respectively (Sandia tests were designed to study synergistic effects). Each of the tests satisfy the intent of IEEE 323-1974. The assemblies were electrically loaded to 20 amperes and 600 volts at the start of the tests. Insulation resistance and capacitance measurements were recorded during the tests to indicate damage.

The equipment assemblies with respect to the sequential and simultaneous tests performed were as follows:

1. Sequential Tests (Two)

Gulton Connector/ITT Cable	1 Assembly
Gulton Connector/Anaconda Cable	1 Assembly
Bendix Connector/ITT Cable	2 Assemblies
ITT Connector/ITT Cable	1 Assembly

2. Simultaneous Test (One)

ITT Connector/ITT Cable	1 Assembly
Bendix Connector/ITT Cable	1 Assembly
Bendix Connector/Anaconda Cable	2 Assemblies

Test Results

Both ITT Cannon connector assemblies and both Gulton connector assemblies showed almost immediate damage in terms of insulation resistance and capacitance as the 70 psig steam was applied.

The ITT Cannon connector assembly failures appeared to be back plane boot seal leakage failures. The assembly construction did not contain potting compound (by design) to protect the pin backs. Therefore, boot failure leads directly to connector failure.

In the case of the Gulton Assemblies, failures were attributed to both the mating surface interface and the back plane seal. The design uses a rigid insert around the mating pins and the O-ring seals are

bypassed by an alignment key slot. This design may lead to leaks due to non-uniform confinement of the O-ring which could cause arcing between pins. Neutron radiography revealed inadequate amounts of potting compound (voids) and cracking of potting compound. These conditions could account for back plane failures. Neutron radiography performed on untested connectors revealed similar conditions, i.e., voids and cracking, thus indicating an apparent quality control problem at Gulton's facility. Other problems detected were identified as:

1. The shrink tube used over the pin cable interface was split lengthwise and had pulled away.

2. The potting material showed virtually no adhesion to, or sealing between, the cable jacket, insulation, and the connector shell.

3. The mechanical clamp had been secured so tightly that it cut the cable jacket.

The Bendix connector assembly was the only type to survive an entire test cycle. One Bendix/Anaconda assembly malfunctioned after about eight days into the 10 psig profile and the Bendix/ITT assembly experienced decreasing resistance and increasing capacitance through the simultaneous tests until both readings were off scale at the end of the 10 psig profile. A second Bendix/Anaconda assembly survived the simultaneous tests. During the sequential tests only Bendix and ITT Cannon assemblies were involved and both assemblies failed. The failures of these assemblies would be difficult to define as either connector or cable failures. The ITT cable exhibited a shrinking characteristic which could have provided a leak path through the sealing medium of the connector.

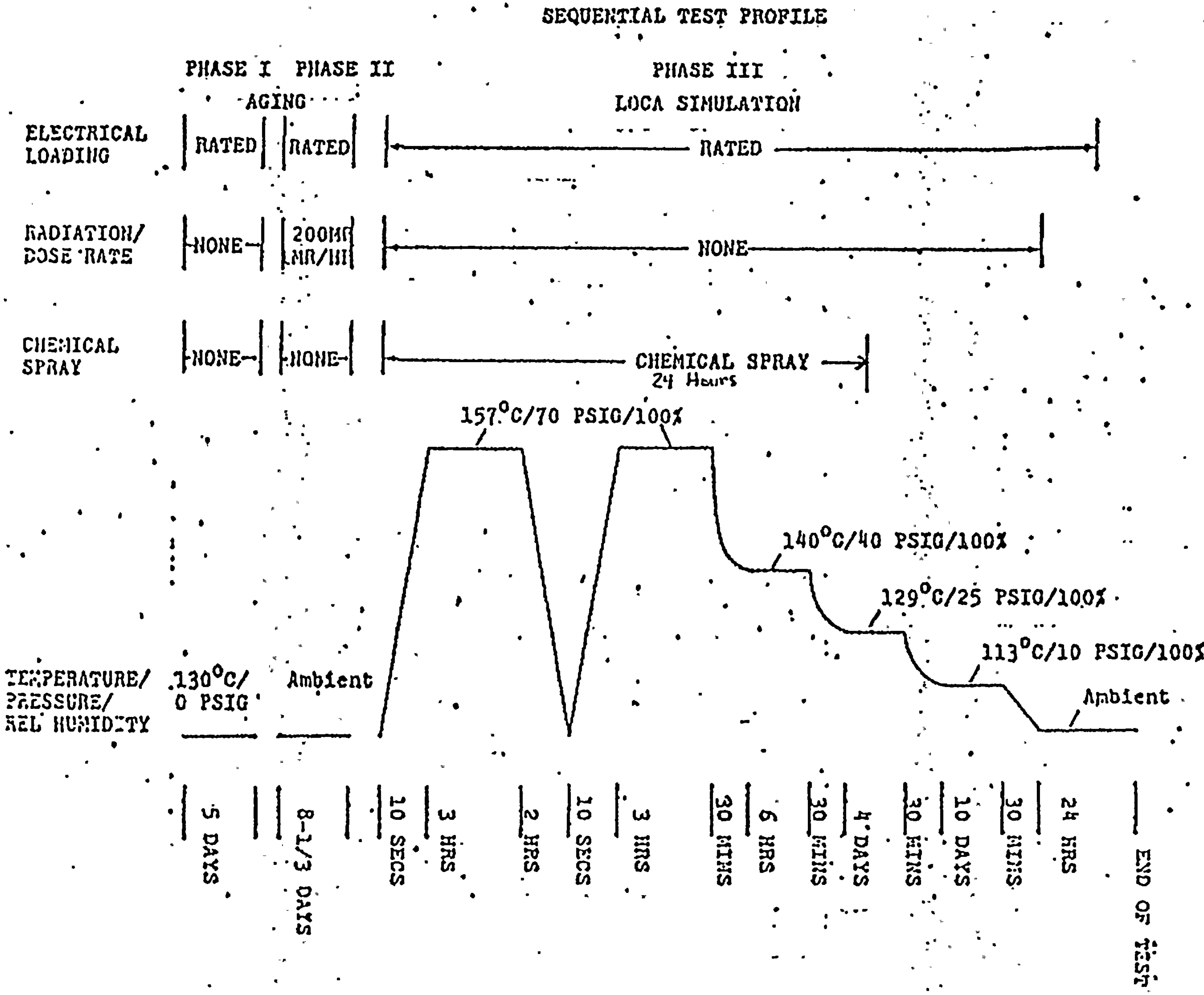
SEQUENTIAL TEST PROFILE
PHASE I PHASE II
AGING
PHASE III
LOCA SIMULATION
ELECTRICAL LOADING
RATED
RATED
RATED
RADIATION/ DOSE RATE
NONE
200MR
MR/HR
NONE
CHEMICAL SPRAY
NONE
NONE
CHEMICAL SPRAY
24 Hours
157°C/70 PSIG/100%
140°C/40 PSIG/100%
129°C/25 PSIG/100%
113°C/10 PSIG/100%
Ambient
TEMPERATURE/ PRESSURE/ REL HUMIDITY
130°C/ 0 PSIG
Ambient
5 DAYS
8-1/3 DAYS
10 SECS
3 HRS
2 HRS
10 SECS
3 HRS
30 MINS
6 HRS
30 MINS
4 DAYS
30 MINS
10 DAYS
30 MINS
24 HRS
END OF TEST

Figure 1

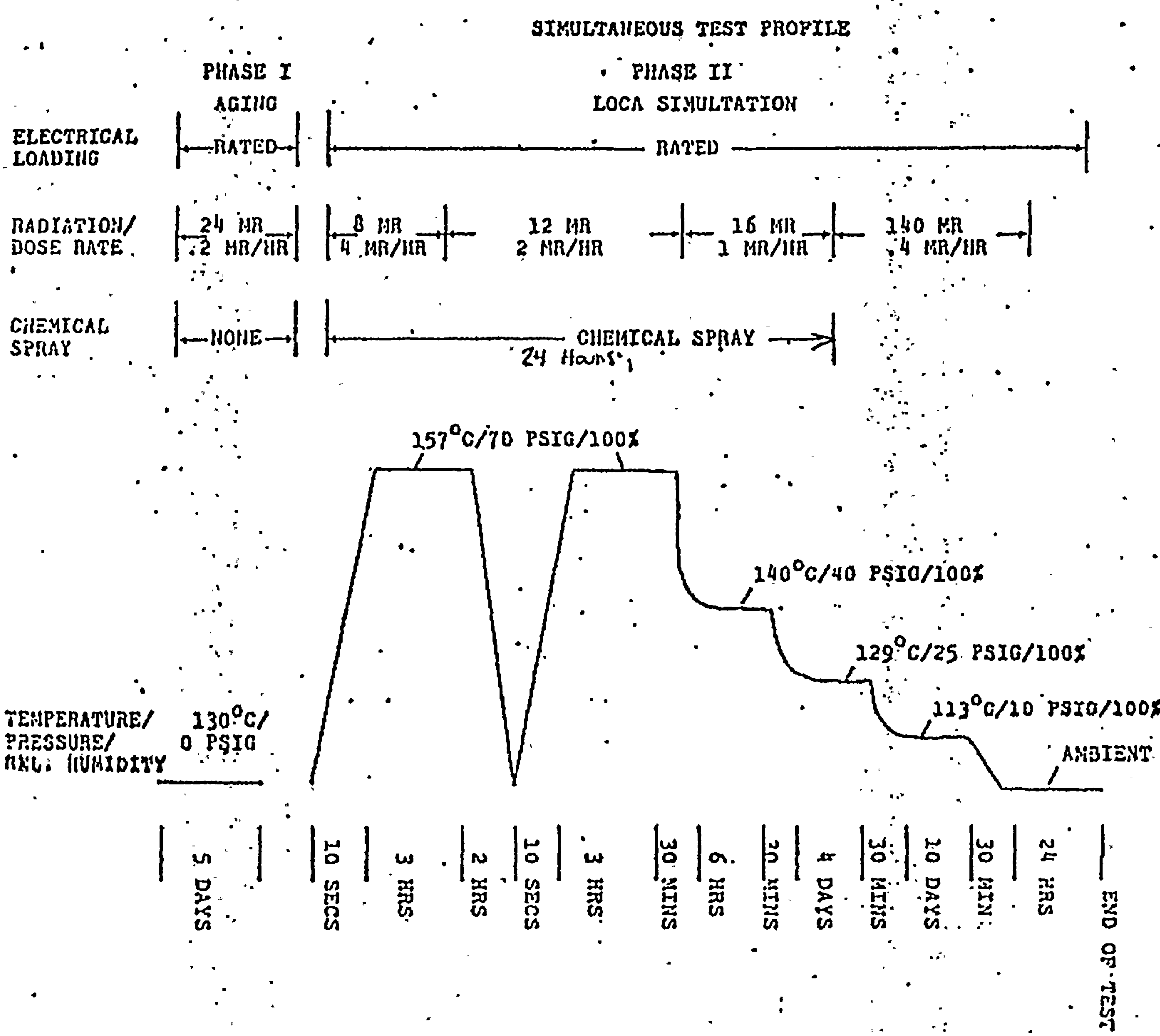
SIMULTANEOUS TEST PROFILE
PHASE I
AGING
PHASE II
LOCA SIMULTATION
ELECTRICAL LOADING
RATED
RATED
RADIATION/ DOSE RATE
24 MR
.2 MR/HR
8 MR
4 MR/HR
12 MR
2 MR/HR
16 MR
1 MR/HR
140 MR
.4 MR/HR
CHEMICAL SPRAY
NONE
CHEMICAL SPRAY
24 Hours
157°C/70 PSIG/100%
140°C/40 PSIG/100%
129°C/25 PSIG/100%
113°C/10 PSIG/100%
AMBIENT
TEMPERATURE/ PRESSURE/ REL. HUMIDITY
130°C/ 0 PSIG
5 DAYS
10 SECS
3 HRS
2 HRS
10 SECS
3 HRS
30 MINS
6 HRS
30 MINS
4 DAYS
30 MINS
10 DAYS
30 MIN.
24 HRS
END OF TEST

Figure 2

[illegible]

[illegible]

IE Bulletin 77-05
November 8, 1977

LISTING OF IE BULLETINS
ISSUED IN 1977

Bulletin No.	Subject	Date Issued	Issued To
77-04	Calculational Error Affecting the Design Performance of a System for Controlling pH of Containment Sump Water Following a LOCA	11/4/77	All PWR Power Reactor Facilities with an Operating License (OL) or Contruction Permit (CP)
77-03	On-Line Testing of the W Solid State Protection System	9/12/77	All W Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
77-02	Potential Failure Mechanism in Certain W AR Relays with Latch Attachments	9/12/77	All Holders of Operating Licenses (OL) or Construction Permits (CP)
77-01	Pneumatic Time Delay Relay Set Point Drift	4/29/77	All Holders of Operating License (OL) or Construction Permit (CP)