October 7, 2005

Xavier Bernard-Bruls
IRS Coordinator
Regulatory Activities Section
Division of Nuclear Installation Safety
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
A-1400 Wien
AUSTRIA

Dear Mr. Bernard-Bruls:

The following operating experience reports from United States reactors are enclosed for your consideration for including in the Advanced Incident Reporting System (AIRS) database:

NRC Information Notice 2005-24: Nonconservatism in Leakage Detection Sensitivity

NRC Information Notice 2005-25: Inadvertent Reactor Trip and Partial Safety Injection Actuation Due to Tin Whisker

NRC Information Notice 2005-26: Results of Chemical Effects Head Loss Tests in a Simulated PWR Sump Pool Environment

Licensee Event Report 2005-004 (Kewaunee): Safe Shutdown Potentially Challenged by Unanalyzed Internal Flooding Events and Inadequate Design

Licensee Event Report 2005-003 (FitzPatrick): Plant Shutdown Due to Through-Wall Crack in Torus

Each report is being submitted in the following two media: (1) a hard copy of the input file for the AIRS database; and (2) a CD containing the input file for the AIRS database in WordPerfect format.

If you have any questions regarding these reports, please contact Brett A. Rini of my staff. He can be reached at 301-415-3931.

Sincerely,

/RA/
Patrick L. Hiland, Chief
Reactor Operations Branch
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Enclosures: As stated

cc w/enclosures:
Dr. Pekka T. Pyy
Administrator, Operating Experience & Human Factors
Nuclear Safety Division
Nuclear Energy Agency
OECD
Le Seine St. Germain, Batiment B
12, Boulevard des Iles
92130 - Issy-les-Moulineaux
FRANCE

If you have any questions regarding these reports, please contact Brett A. Rini of my staff. He can be reached at 301-415-3931.

Sincerely,

/RA/
Patrick L. Hiland, Chief
Reactor Operations Branch
Division of Inspection Program Management
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Dr. Pekka T. Pyy
Administrator, Operating Experience & Human Factors
Nuclear Safety Division
Nuclear Energy Agency
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Le Seine St. Germain, Batiment B
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92130 - Issy-les-Moulineaux
FRANCE

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ADAMS ACCESSION NUMBER: PACKAGE: ML052790250, MEMO, ML052790256, ATTACHMENTS ML052590238, ML051440302, ML052510120

OFFICE	OES:IROB:DIPM	TL:OES:IROB:DIPM	SC:OES:IROB:DIPM	C:IROB:DIPM
NAME	BRini	IJung	MRoss-Lee	PLHiland
DATE	10/07/2005	10/06/2005	10/07/2005	10/07/2005

OFFICIAL RECORD COPY

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	N/A	DATE RECEIVED

EVENT TITLE

NRC Information Notice 2005-24: Nonconservatism in Leakage Detection Sensitivity

COUNTRY	PLANT AND UNIT	REACTOR TYPE
United States	Many	Generic
INITIAL STATUS	**RATED POWER (MWe NET)**	
N/A	N/A	
DESIGNER	**1st COMMERCIAL OPERATION**	
N/A	N/A	

ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to inform addressees that the reactor coolant activity assumptions for containment radiation gas channel monitors may be nonconservative. As a result, the containment gas channel may not be able to detect a 1 gallon-per-minute (1-gpm) leak within 1 hour. It is expected that the recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

NRC INFORMATION NOTICE 2005-24

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

---

1. **Reporting Categories:**
   1.2.6
   1.4

2. **Plant Status Prior to the Event:**
   2.0

3. **Failed/Affected Systems:**
   3.IG
   3.IH

4. **Failed/Affected Components:**
   4.1.5

5. **Cause of the Event:**
   5.1.5.0
   5.5.7

6. **Effects on Operation:**
   6.0

7. **Characteristics of the Incident:**
   7.0

8. **Nature of Failure or Error:**
   8.1
   8.4

9. **Nature of Recovery Actions:**
   9.1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

August 3, 2005

NRC INFORMATION NOTICE 2005-24: NONCONSERVATISM IN LEAKAGE DETECTION SENSITIVITY

ADDRESSEES

All holders of operating license or construction permits for nuclear power reactors, except those that have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to inform addressees that the reactor coolant activity assumptions for containment radiation gas channel monitors may be nonconservative. As a result, the containment gas channel may not be able to detect a 1 gallon-per-minute (1-gpm) leak within 1 hour. It is expected that the recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

Several nuclear power plant licensees have reported problems with the detection capabilities of containment radiation gas channel monitors. The following gives several examples of these reports.

On May 2, 2005, the McGuire nuclear power plant licensee reported that the containment atmosphere radioactivity monitors were not sensitive enough for their intended function of detecting a 1-gpm reactor coolant system (RCS) leak within 1 hour (Licensee Event Report (LER) 50-369/2005-01, ADAMS Accession No. ML051310167). This resulted in a Severity Level IV noncited violation.

The McGuire licensee declared the atmosphere monitors inoperable and performed compensatory actions in accordance with plant technical specifications. The compensating actions were to (1) establish temporary alarm setpoints to provide earlier notification should a significant RCS leak occur, (2) instruct operators on other methods of RCS leak detection, (3) establish sensitivities as low as practical based on actual RCS radioactivity levels, (4) periodically review the sensitivities for revision as needed, (5) provide additional training as needed, and (6) consider submitting a license amendment request to clarify the capabilities of the leak detection instrumentation.

ML051780073

In February 2005, NRC inspectors at the Catawba nuclear power plant identified a noncited violation of Technical Specification 5.4.1.a, “Written Procedures,” because the licensee failed to establish and maintain an adequate procedure for the required containment atmosphere radioactivity monitor surveillance in that the associated alarm function was not set or tested to alarm at a value equivalent to 1 gpm in 1 hour for a realistic current reactor coolant activity level (NRC Integrated Inspection Report 50-413/2005-02 and 50-414/2005-02, ADAMS Accession No. ML051160367).

The Catawba licensee also declared these channels to be inoperable and is performing compensatory actions in accordance with plant technical specifications.

In June 2003, an NRC inspection made a similar finding at Callaway (NRC Inspection Report 50-483/2003-04, ADAMS Accession No. ML032020562) that resulted in a noncited violation. The gas channel monitor was not capable of performing its design basis function of detecting a 1 gpm RCS leak within 1 hour. The calculation for the gas channel monitor response used an RCS source term corresponding to an assumed 0.1 percent failed fuel but, because of improved fuel performance and RCS chemistry control, the plant operated with an RCS source term several orders of magnitude smaller.

The Callaway licensee responded to this situation similarly by (1) declaring the gas channel out of service to prevent its being credited for leakage detection and (2) considering a license amendment request to revise the final safety analysis report and technical specification bases to reflect actual leakage detection capabilities.

DISCUSSION

The NRC requires licensees to use a means of detecting and, to the extent practical, identifying the location of any sources of RCS leakage (Title 10 of the Code of Federal Regulations, Part 50, Appendix A, “General Design Criteria [GDC] for Nuclear Power Plants,” Criterion 30, “Quality of Reactor Coolant Pressure Boundary”). The NRC provided guidance on meeting GDC 30 in Regulatory Guide (RG) 1.45, “Reactor Coolant Pressure Boundary Leakage Detection Systems.” Some licensees committed to using RG 1.45 as the basis for meeting GDC 30.

RG 1.45 states that an acceptable means would provide for adequate sensitivity and response time of all leakage detection systems to detect a leakage rate of 1 gpm in less than 1 hour. Further, the acceptable means would employ at least three separate detection methods. Two of these methods are monitoring sump level and sump flow and monitoring airborne particulate radioactivity. The third method is either monitoring the condensate flow rate from air coolers or monitoring airborne gaseous radioactivity. The guide also states that a “realistic” primary radioactivity concentration should be assumed when analyzing the sensitivity of leak detection systems.

During original plant licensing, the typical calculation for the technical specification for gas channel monitor response used an RCS source term corresponding to an assumed 0.1 percent failed fuel. Nowadays, because of improvements in fuel performance and RCS chemistry control, the actual RCS source term can be orders of magnitude smaller. Though desirable, a

small source term can result in reduced leakage monitoring capabilities. Using a realistic RCS source term, a 1 gpm RCS leak would likely not be detected by a gas channel monitor for a much greater time than within 1 hour. The 0.1-percent failed fuel assumption introduces a nonconservatism into the technical specifications. Guidance on resolving such a nonconservatism is given in NRC Administrative Letter 98-10, “Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety.”
The consistency of leakage detection systems with RG 1.45 has been questioned at several nuclear power plants. See NUREG/CR-6861, “Barrier Integrity Research Program,” December 2004 (ADAMS Accession No. ML043580207) for a good discussion of detector sensitivities.

This information notice requires no specific action or written response. Please direct any questions about this matter to the technical contact(s) listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/ By David C. Trimble Acting For/
Patrick L. Hiland, Chief
Reactor Operations Branch
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Technical Contacts: Michael Peck, RIV
573-676-3181
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Vernon Hodge, NRR
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E-mail: cvh@nrc.gov

Note: NRC generic communications may be found on the NRC public Website, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	04/17/2005	DATE RECEIVED

EVENT TITLE

NRC Information Notice 2005-25: Inadvertent Reactor Trip and Partial Safety Injection Actuation Due to Tin Whisker

COUNTRY	PLANT AND UNIT	REACTOR TYPE
US	Millstone 3	PWR
INITIAL STATUS	**RATED POWER (MWe NET)**	
Full Power	1131	
DESIGNER	**1st COMMERCIAL OPERATION**	
Westinghouse	4/23/1986	

ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees about recent operating experience related to the growth of “tin whiskers” in electronic circuits at nuclear power stations. Recipients are expected to review the information for applicability to their facilities and consider appropriate actions to avoid similar problems. However, the measures suggested in this information notice are not NRC requirements and no specific action or written response is required.

NRC INFORMATION NOTICE 2005-25

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

---

1. **Reporting Categories:**
   1.4
   1.6

2. **Plant Status Prior to the Event:**
   2.1.1

3. **Failed/Affected Systems:**
   3.BG
   3.IN

4. **Failed/Affected Components:**
   4.3.8

5. **Cause of the Event:**
   5.1.2.5
   5.7.2

6. **Effects on Operation:**
   6.1.1
   6.4

7. **Characteristics of the Incident:**
   7.0

8. **Nature of Failure or Error:**
   8.1

9. **Nature of Recovery Actions:**
   9.2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

August 25, 2005

NRC INFORMATION NOTICE 2005-25: INADVERTENT REACTOR TRIP AND PARTIAL SAFETY INJECTION ACTUATION DUE TO TIN WHISKER

ADDRESSEES

All holders of operating licenses for pressurized-water reactors (PWRs) and boiling-water reactors (BWRs) except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees about recent operating experience related to the growth of “tin whiskers” in electronic circuits at nuclear power stations. Recipients are expected to review the information for applicability to their facilities and consider appropriate actions to avoid similar problems. However, the measures suggested in this information notice are not NRC requirements and no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

On April 17, 2005, Millstone Nuclear Generating Station, Unit 3, experienced an unexpected safety injection actuation and reactor trip caused by a fault on a solid state protection system (SSPS) circuit card. The fault generated a false low steamline pressure signal, bypassing the 2-out-of-3 SSPS logic and causing the A safety train actuation and reactor trip. The licensee examined the failed circuit card using a magnifying glass and found a microscopic tin filament (approximately 2 mm long). The filament created a bridge between the affected diode and the output trace on the card. This microscopic filament of tin called “tin whisker,” had grown out of the tin coating covering the leads of the diode.

The licensee inspected all circuit cards in the SSPS and discovered tin whiskers on other circuit cards. In each case, the whisker appeared to originate at the tin coating on diode leads. Suspect cards were either replaced or cleaned before being placed back in service. The licensee sampled additional circuit cards from other important plant systems but found no other evidence of tin whiskers.

BACKGROUND

Tin whiskers are electrically conductive crystalline structures of tin that sometimes grow from surfaces where pure tin (especially electroplated tin) is used as a final finish. Tin whiskers have been observed to grow to lengths of several millimeters (mm) and in rare instances up to 10 mm. Electronic system failures have been attributed to short circuits caused by tin whiskers that bridge closely spaced circuit elements maintained at different electrical potentials.

Tin whiskers appear to have increased following international efforts to remove alloying metals such as lead from solder and other circuit card manufacturing materials to reduce environmental and health hazards. With the move toward lead-free electronics, tin has become a drop-in replacement for the tin-lead finish currently used for electrical component terminations. The move to lead-free electronics means that failures of some high-reliability components may continue to increase until a solution to the tin whiskers problem is found. Tin whiskers have been cited as the cause for various minor component failures in the nuclear industry and significant failures in the aerospace industry.

DISCUSSION

Some of the failures due to whiskers are documented in licensee event reports (LERs):

Plant	LER No.	NUDOCS Accession No.
Dresden Unit 2	50-237/1987-22	8709230145
Duane Arnold	50-331/1990-04	9005010072
Dresden Unit 2	50-237/1997-19	9801270112
South Texas Unit	50-499/1999-06	9910080186

In most of the events, metallic whiskers caused a short of the local power range monitors (LPRM) detectors resulting in a momentary spike on the average power range monitors (APRMs). In other cases, whiskers resulted in a failure of a channel input relay to the engineered safety features (ESF) actuation logic. In most cases, failure of the channel inputs in to the reactor protection system (RPS) or the ESF actuation did not result in a full RPS or ESF actuation. Only half of the RPS or ESF logic was met.

The incident at Millstone Unit 3 demonstrates that a single tin whisker can cause a protective feature to actuate. It is reasonable to assume that the same phenomenon could also prevent a protective system actuation. The extent-of-condition review performed at Millstone also showed that circuit cards need not be in service to be susceptible to whiskering. Research available from NASA's Goddard Space Center (http://nepp.nasa.gov/whisker) and Computer Aided Life Cycle Engineering (CALCE) at the University of Maryland supports this discovery and provides other valuable information on prevention techniques and growth mechanisms. While the information provided directly states that the exact mechanism for growth is unknown, common growth conditions and theories are discussed.

The data from the extent-of-condition review at Millstone Unit 3, NASA and CALCE information indicate that more than one manufacturer makes high-reliability circuit cards susceptible to tin whiskering. The data also indicates that tin whiskering is not significantly influenced by the environment in which the cards are used. Therefore, if one card procured from a specific vendor shows evidence of whiskering, all cards of that type from the same manufacturer can be expected to show signs of whiskering. In general, components containing 3% or greater lead concentration in the solder and/or manufactured with conformal coatings appear to be less susceptible to tin whiskering.

CONTACTS

This information notice requires no specific action or written response. Please direct any questions about this matter to the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/
Patrick L. Hiland, Chief
Reactor Operations Branch
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Technical Contacts: Paul Rebstock, NRR
301-415-3295
E-mail: pjr1@nrc.gov

Thomas Sicola, R-I/DRS
610-337-5109
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Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	N/A	DATE RECEIVED

EVENT TITLE

NRC Information Notice 2005-26: Results of Chemical Effects Head Loss Tests in a Simulated PWR Sump Pool Environment

COUNTRY	PLANT AND UNIT	REACTOR TYPE
US	Generic	PWR

INITIAL STATUS	RATED POWER (MWe NET)
N/A	N/A

DESIGNER	1st COMMERCIAL OPERATION
N/A	N/A

ABSTRACT

The U.S. Nuclear Regulatory Commission is issuing this information notice to inform addressees about recent NRC-sponsored research results related to head loss from chemical effects in a simulated PWR sump pool environment. The NRC anticipates that recipients will review the information for applicability to their facilities and consider taking actions, as appropriate, to avoid similar issues. However, no specific action or written response is required.

NRC INFORMATION NOTICE 2005-26

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

---

1. **Reporting Categories:**
   1.2.5
   1.4

2. **Plant Status Prior to the Event:**
   2.0

3. **Failed/Affected Systems:**
   3.BG
   3.SA

4. **Failed/Affected Components:**
   4.2.8

5. **Cause of the Event:**
   5.1.1.8
   5.1.3.1

6. **Effects on Operation:**
   6.0

7. **Characteristics of the Incident:**
   7.5

8. **Nature of Failure or Error:**
   8.3

9. **Nature of Recovery Actions:**
   9.0

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

September 16, 2005

NRC INFORMATION NOTICE 2005-26: RESULTS OF CHEMICAL EFFECTS HEAD LOSS TESTS IN A SIMULATED PWR SUMP POOL ENVIRONMENT

ADDRESSEES

All holders of operating licenses for pressurized water reactors (PWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.

PURPOSE

The U.S. Nuclear Regulatory Commission is issuing this information notice to inform addressees about recent NRC-sponsored research results related to head loss from chemical effects in a simulated PWR sump pool environment. The NRC anticipates that recipients will review the information for applicability to their facilities and consider taking actions, as appropriate, to avoid similar issues. However, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

Generic Safety Issue (GSI) 191 addresses the potential for debris accumulation on PWR sump screens to affect emergency core cooling system (ECCS) pump net positive suction head margin. The NRC has issued Bulletin 2003-01, “Potential Impact of Debris Blockage On Emergency Sump Recirculation At Pressurized Water Reactors,” and Generic Letter (GL) 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents At Pressurized Water Reactors,” related to the GSI-191 resolution.
GL 2004-02 requests, in part, that licensees evaluate the maximum head loss postulated from debris accumulation (including chemical effects) on the submerged sump screen. Chemical effects are corrosion products, gelatinous material, or other chemical reaction products that form as a result of interaction between the PWR containment environment and containment materials after a loss-of-coolant accident (LOCA). NRC and the nuclear industry jointly developed an integrated chemical effects test (ICET) program to determine if chemical reaction products can form in representative PWR post-LOCA containment sump environments. These tests were conducted by Los Alamos National Laboratory at the University of New Mexico. The ICET series involved five tests, each representing a different subset of expected post-LOCA environments within existing PWR plants. Although chemical products were observed in all of

the ICET environments, the head loss associated with these products was not evaluated as it was outside the scope of the ICET program. NRC initiated additional testing to obtain some insights on the head loss associated with chemical products that may form in PWR sump pools.

Head loss testing is being performed at the Argonne National Laboratory. Initial testing has been done in a piping loop containing a simulated sump pool environment intended to represent the ICET Test 3 conditions. ICET Test 3 was performed in a borated water environment containing trisodium phosphate (TSP), various metallic and non-metallic sample coupons representative of containment materials, and a mixture of insulation (80% calcium silicate, 20% fiberglass) samples. This environment was selected for initial head loss testing based on the early formation of chemical product during ICET Test 3 and the characteristics of this product observed during and after this test (NRC ADAMS Package Accession Number ML052140490). During initial testing to simulate these observed products, significant head loss was measured across a test screen containing a preexisting fiber bed. The Argonne tests and initial test results are described in detail in the attachment, “Chemical Effects/Head Loss Testing Quick Look Report, Tests 1 and 2,” dated September 16, 2005.

DISCUSSION

As part of the GL 2004-02 response, licensees are required to evaluate the sump screen head loss consequences of any chemical effects in an integrated manner with other postulated post-LOCA conditions. These recent research results indicate that a simulated sump pool environment containing phosphate and dissolved calcium can rapidly produce a calcium phosphate precipitate that, if transported to a fiber bed covered screen, produces significant head loss. The attachment report contains several interesting observations:

- Significant head loss was observed in tests combining TSP with a higher concentration of dissolved calcium (simulating the ICET Test 3 environment) and in tests with TSP and lower dissolved calcium concentrations (i.e., less than the ICET 3 environment).
- Small-scale leaching tests were done with calcium silicate insulation. The amount of calcium that will dissolve appears to depend more on the initial pH of the solution than on the amount of calcium-silicate insulation placed into solution. Lower initial pH solutions produced greater amounts of dissolved calcium.
- The amount of calcium phosphate precipitant in an ICET Test 3 type environment may be limited by the amount of phosphate available from the TSP.

This information is relevant to plants containing phosphate (e.g., plants using TSP as a sump pool buffering agent) and calcium sources (e.g., insulation, concrete) that may dissolve within the post-LOCA containment pool with sufficient concentrations to form calcium phosphate precipitate. These test results indicate that substantial head loss can occur if sufficient calcium phosphate is produced in a sump pool and transported to a preexisting fiber bed on the sump screen.

Although significant increases in head loss were observed due to chemical effects in these tests, it is important to note that these head loss results were obtained in a recirculating test loop not intended to be prototypical of a PWR plant containment. For example, the calcium phosphate precipitant was formed by introducing calcium chloride into a TSP buffered solution immediately upstream and at a higher elevation than a screen with a preestablished fiber bed. The test loop orientation and method of calcium introduction result in transport of virtually all chemical products to the fiber bed covered screen. Parameters that may influence head loss in these tests include screen approach velocity, fiber bed thickness, relative arrival times for debris and chemical precipitates, and loop fluid recirculation time. Applicability of these results to plant specific environments may also be affected by these and other variables (e.g., insulation materials, break location, and sump design).

The NRC is continuing head loss testing in simulated PWR sump pool environments that use other chemical species to buffer pH.

CONTACTS

This information notice does not require any specific action or written response. Please direct any questions about this matter to the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/
Patrick L. Hiland, Chief
Reactor Operations Branch
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Technical Contacts: Paul Klein, NRR
301-415-4030
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Robert Tregoning, RES
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Attachment: Chemical Effects/Head-Loss Testing Quick Look Report, Tests 1 and 2

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE N/A	DATE RECEIVED

EVENT TITLE

Licensee Event Report 305-2005-004 (Kewaunee): Safe Shutdown Potentially Challenged By Unanalyzed Internal Flooding Events and Inadequate Design (ML051440302)

COUNTRY	PLANT AND UNIT	REACTOR TYPE
US	Kewaunee	PWR

INITIAL STATUS	RATED POWER (MWe NET)
Refueling	539

DESIGNER	1st COMMERCIAL OPERATION
Westinghouse	6/16/1974

ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) is submitting this publically available Licensee Event Report (LER) to inform IRS users of an event that was provisionally rated INES Level 2. The accuracy and completeness of this LER has not been confirmed by the NRC. This is a preliminary report, and the NRC will submit a final report once the relevant Information Notice has been completed.

Licensee Abstract:
On March 15, 2005 with the plant in Refueling Shutdown Mode, Nuclear Management Company (NMC) personnel determined that the Kewaunee plant design for protection against internal flooding would not ensure that required equipment would be protected from the postulated failure of non-safety related piping in the turbine building. High water level in the turbine building would result in water flowing into certain Engineered Safety Features equipment rooms. Documentation which considers specific flooding events from postulated failures of plant equipment exists, however, a complete internal plant flooding analysis was not developed during or subsequent to the plant's original design. In response to inadequate plant design, physical changes are being made to minimize challenges to plant equipment and personnel in combating potential flooding events. Analysis continues to determine the potential for and effects of flooding events occurring, and to enhance and document the plant's design for internal flooding. Although this LER is not associated with an event resulting in actual flooding of any portion of the plant, the potential for certain piping and tank failures resulting in unacceptable flooding exists. A past operability evaluation is underway to assess what equipment would have failed during postulated flooding events. The Significance Determination Process will be used to assess the safety consequences and implications for any equipment that would have failed. This information will be addressed in a supplement to this LER. This report does not involve a safety system functional failure.

Licensee Event Report 305-2005-004 (Kewaunee)

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

---

1. **Reporting Categories:**
   1.2.6
   1.3.1
   1.4

2. **Plant Status Prior to the Event:**
   2.3.2

3. **Failed/Affected Systems:**
   3.E
   3.IP
   3.SG

4. **Failed/Affected Components:**
   4.3

5. **Cause of the Event:**
   5.1.6.4
   5.7.1

6. **Effects on Operation:**
   6.0

7. **Characteristics of the Incident:**
   7.13

8. **Nature of Failure or Error:**
   8.3

9. **Nature of Recovery Actions:**
   9.1

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	N/A	DATE RECEIVED

EVENT TITLE

Licensee Event Report 333-2005-003 (FitzPatrick): Plant Shutdown Due to Through-Wall Crack in Torus (ML052510120)

COUNTRY	PLANT AND UNIT	REACTOR TYPE
US	FitzPatrick	BWR

INITIAL STATUS	RATED POWER (MWe NET)
Full Power	825

DESIGNER	1st COMMERCIAL OPERATION
GE	7/28/1975

ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) is submitting this publically available Licensee Event Report (LER) to inform IRS users of an event that was provisionally rated INES Level 2. The accuracy and completeness of this LER has not been confirmed by the NRC. This is a preliminary report, and the NRC will submit a final report once the relevant Information Notice has been completed.

Licensee Abstract:
On June 27, 2005, while the plant was operating at 100 percent power, inspectors discovered a small leak (1 -2 drops/minute) through the Torus shell. A subsequent engineering evaluation determined that the Primary Containment was inoperable, the plant was shutdown in accordance with normal shutdown procedures. The declaration of containment inoperability resulted in entering the site emergency plan and declaring an Unusual Event (UE).

The plant's High Pressure Coolant Injection (HPCI) exhaust line, although consistent with original GE design specifications, does not include a HPCI turbine exhaust line sparger. A properly designed sparger is expected to reduce local Torus shell stresses resulting from HPCI turbine exhaust pressure pulses. The plant did not install a HPCI sparger due to inadequate information transfer to FitzPatrick from General Electric (GE) and other nuclear facilities regarding concerns with pressure instability and vibration inside the Torus.

As a result of this design deficiency, the Torus shell experienced localized stress, high cycle fatigue due to rapid condensation of the HPCI exhaust steam at the ring girder weld heat affected zone, resulting in a Torus shell through-wall crack.

As part of the corrective actions, an American Society of Mechanical Engineers (ASME) Code repair was performed to repair the Torus shell. In addition, the HPCI exhaust line and ring girder gusset attachment will be modified as required to reduce the associated stresses.

There were no actual safety consequences associated with this event.

Licensee Event Report 333-2005-003 (FitzPatrick)

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

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1. **<u>Reporting Categories</u>:**
   1.2.3
   1.4

2. **<u>Plant Status Prior to the Event</u>:**
   2.1.1

3. **<u>Failed/Affected Systems</u>:**
   3.SA

4. **<u>Failed/Affected Components</u>:**
   4.2.5

5. **<u>Cause of the Event</u>:**
   5.1.1.3
   5.1.1.6
   5.1.1.7

6. **<u>Effects on Operation</u>:**
   6.2

7. **<u>Characteristics of the Incident</u>:**
   7.3

8. **<u>Nature of Failure or Error</u>:**
   8.1

9. **<u>Nature of Recovery Actions</u>:**
   9.3