August 8, 2001

Mr. Valeri Tolstykh
Regulatory Activities Unit
Safety Assessment Section
Division of Nuclear Installation Safety
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100, A-1400
Vienna, Austria

Dear Mr. Tolstykh:

Enclosed are the following IRS reports:

- NEGLECTED FIRE EXTINGUISHER MAINTENANCE CAUSES FATALITY (NRC Information Notice 2001-04)

- THROUGH-WALL CIRCUMFERENTIAL CRACKING OF REACTOR PRESSURE VESSEL HEAD CONTROL ROD DRIVE MECHANISM PENETRATION NOZZLES AT OCONEE NUCLEAR STATION, UNIT 3 (NRC Information Notice 2001-05)

- CENTRIFUGAL CHARGING PUMP THRUST BEARING DAMAGE NOT DETECTED DUE TO INADEQUATE ASSESSMENT OF OIL ANALYSIS RESULTS AND SELECTION OF PUMP SURVEILLANCE POINTS (NRC Information Notice 2001-06)

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 3.5-inch HD diskette containing the input file for the AIRS database in Microsoft Word 6.0 format.

If you have any questions regarding these reports, please call Edward F. Goodwin, of my staff. He can be reached at (301) 415-1154.

Sincerely,

/RA/

John R. Tappert, Acting Chief
Operational Experience and
Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Enclosures: As stated

cc w/enclosures 1 and 2:
Mr. Lennart Carlsson
Nuclear Safety Division
Nuclear Energy Agency
Organization for Economic
Cooperation and Development
Le Seine Saint Germain
12, Boulevard des Iles
92130, Issy-les-Moulineaux, France

Mr. Valeri Tolstykh
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Safety Assessment Section
Division of Nuclear Installation Safety
International Atomic Energy Agency
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cc w/enclosures 1 and 2:
Mr. Lennart Carlsson
Nuclear Safety Division
Nuclear Energy Agency
Organization for Economic
Cooperation and Development
Le Seine Saint Germain
12, Boulevard des Iles
92130, Issy-les-Moulineaux, France

Accession No.: ML012190152

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INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	2000/8/25	

EVENT TITLE

NEGLECTED FIRE EXTINGUISHER MAINTENANCE CAUSES FATALITY
(NRC Information Notice 2001-04)

COUNTRY	PLANT AND UNIT	REACTOR TYPE
USA	Generic	N/A

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

DESIGNER	1st COMMERCIAL OPERATION
N/A	N/A

ABSTRACT

This IRS report discusses the danger of corrosion to fire extinguishers.

NEGLECTED FIRE EXTINGUISHER MAINTENANCE CAUSES FATALITY
(NRC Information Notice 2001-04)

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		
2.	Plant Status Prior to the Event:	2.0		
3.	Failed/Affected Systems:	3.KP		
4.	Failed/Affected Components:	4.0		
5.	Cause of the Event:	5.1.1.1		
6.	Effects on Operation:	6.0		
7.	Characteristics of the Incident:	7.12		
8.	Nature of Failure or Error:	8.0		
9.	Nature of Recovery Actions:	9.0		

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
Washington, D.C. 20555-0001

April 11, 2001

NRC INFORMATION NOTICE 2001-04: NEGLECTED FIRE EXTINGUISHER MAINTENANCE CAUSES FATALITY

Addressees

All holders of licenses for nuclear power, research, and test reactors and fuel cycle facilities.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to the danger of corrosion to fire extinguishers. It is expected that recipients will review the information for applicability and consider actions, as appropriate, to ensure safety at their facilities. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

On August 25, 2000, at the Port of Rotterdam, Netherlands, an employee who was trying to extinguish a small fire activated a fire extinguisher, which in-turn exploded, killing the employee. The cause of the explosion was corrosion under a rubber or plastic base protecting the bottom of the extinguisher. This base had trapped moisture next to the shell of the extinguisher accelerating corrosion. The corrosion was hidden by the flange and went unnoticed during inspections. The extinguisher was manufactured in 1987 by Ansul, Belgium, which is not affiliated with Ansul Incorporated, USA.

Following the incident, a number of other extinguishers were checked and other cases of serious corrosion were found. The vendor had conducted maintenance for the first few years after purchase of the extinguishers, but another contractor had been doing the periodic maintenance for the last nine years. The vendor has distributed a warning to owners of these extinguishers saying the annual inspection must include a visual inspection of the extinguishers with the base removed.

A warning was issued on this incident by the Government Industry Data Exchange Program, Agency Action Notice ANN-U-01-02 on October 5, 2000. This notice includes pictures of a corroded cylinder and is posted at the following Web site:
http://www.msha.gov/ALERTS/equipment/ansuldrychem.pdf

Discussion

The NRC endorses the use of the National Fire Protection Associations' (NFPA) Standard for Portable Fire Extinguishers, NFPA 10. The standard provides guidance for the selection, installation, design, inspection, and maintenance of portable fire extinguishers. A general requirement is that extinguishers installed in an environment where they may be subjected to physical damage or degradation should be adequately protected. This includes wet areas which are conducive to corrosion (cooling towers, intake pumping stations, utility vehicles, etc.). During monthly inspections, visual examination of extinguishers should check for obvious physical damage, such as corrosion, leakage, and denting. If damage is detected, the extinguisher should be removed from service and given applicable maintenance. NFPA 10 requires fire extinguisher maintenance to be conducted at least annually and some extinguishers get an internal as well as an external examination. In addition to annual maintenance, hydrostatic testing is required every 5 to 12 years, depending on the type of extinguisher. Extinguishers that fail to pass visual examination or hydrostatic tests are marked "CONDEMNED" and should never be reused. It should also be noted that fire extinguishers are pressure vessels and some facilities elect to use a fire equipment servicing contractor to maintain and recharge their fire extinguishers.

Conclusion

Fire extinguishers are often the first line of defense in fire suppression, and should be readily available to suppress a fire in its incipient stages. Fire extinguishers should not constitute a hazard to the personnel and property they are designed to protect. Proper installation, inspection, and maintenance by qualified personnel should ensure fire extinguishers have a long service life.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or your facility's project manager.

/RA/
Ledyard B. Marsh, Chief
Events Assessment, Generic Communications
and Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

/RA/
Michael F. Weber, Director
Division of Fuel Cycle Safety
and Safeguards
Office of Nuclear Material Safety
and Safeguards

Technical Contacts:

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Attachments:
1. List of Recently Issued NRC Information Notices
2. List of Recently Issued NMSS Information Notices

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	2001/02/18	

EVENT TITLE

THROUGH-WALL CIRCUMFERENTIAL CRACKING OF REACTOR PRESSURE VESSEL HEAD CONTROL ROD DRIVE MECHANISM PENETRATION NOZZLES AT OCONEE NUCLEAR STATION, UNIT 3 (NRC Information Notice 2001-05)

COUNTRY	PLANT AND UNIT	REACTOR TYPE
USA	Generic	PWR

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

DESIGNER	1st COMMERCIAL OPERATION
B&W	N/A

ABSTRACT

This IRS report discusses the recent detection of through-wall circumferential cracks in two of the control rod drive mechanism penetration nozzles and weldments at the Oconee Nuclear Station, Unit 3.

THROUGH-WALL CIRCUMFERENTIAL CRACKING OF REACTOR PRESSURE VESSEL HEAD CONTROL ROD DRIVE MECHANISM PENETRATION NOZZLES AT OCONEE NUCLEAR STATION, UNIT 3 (NRC Information Notice 2001-05)

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

1.	**Reporting Categories:**	1.2.2		
2.	**Plant Status Prior to the Event:**	2.3		
3.	**Failed/Affected Systems:**	3.AB		
4.	**Failed/Affected Components:**	4.2.10		
5.	**Cause of the Event:**	5.1.1.1		
6.	**Effects on Operation:**	6.0		
7.	**Characteristics of the Incident:**	7.2		
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. 20555-0001

April 30, 2001

NRC INFORMATION NOTICE 2001-05: THROUGH-WALL CIRCUMFERENTIAL CRACKING OF REACTOR PRESSURE VESSEL HEAD CONTROL ROD DRIVE MECHANISM PENETRATION NOZZLES AT OCONEE NUCLEAR STATION, UNIT 3

Addressees

All holders of operating licenses for pressurized water nuclear power reactors except those who have 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 alert addressees to the recent detection of through-wall circumferential cracks in two of the control rod drive mechanism (CRDM) penetration nozzles and weldments at the Oconee Nuclear Station, Unit 3 (ONS3).

It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific actions or written response is required.

Description of Circumstances

On February 18, 2001, with ONS3 in Mode 5, Duke Energy Corporation (the licensee) performed a visual examination (VT-2) of the outer surface of the unit's reactor pressure vessel (RPV) head to inspect for indications of borated water leakage. This RPV head inspection was performed as part of a normal surveillance during a planned maintenance outage. The VT-2 revealed the presence of small amounts of boric acid residue in the vicinity of nine of the 69 CRDM penetration nozzles (Figures 1 and 2). Subsequent nondestructive examinations (NDEs) identified 47 recordable crack indications in these nine degraded CRDM penetration nozzles. The licensee initially characterized these flaws as either axial or below-the-weld circumferential indications, and initiated repairs of the degraded areas. NDEs of nine additional CRDM penetration nozzles from the same heat of material were conducted for "extent of condition" purposes, but did not detect recordable indications.[1]

ML011160588

[1] Axial flaws are flaws that propagate along the inside or outside diameter length of the CRDM nozzle. Below-the-weld circumferential indications are apparent flaws oriented around the circumference of the nozzle, beneath the RPV head and below the area where the nozzle is welded to the RPV head. A recordable indication is one that exceeds the NDE acceptance criteria.

Subsequent dye-penetrant testing (PT) revealed additional indications in two of the nine degraded penetration nozzles. While affecting further repairs of these indications, the licensee identified that each nozzle had significant circumferential cracks in the nozzle above the weld. Further investigations and metallurgical examinations revealed that these cracks had initiated from the outside diameter (OD) of the CRDM penetration nozzles. The circumferential crack in the #56 CRDM nozzle was through-wall, and the #50 nozzle had pin hole through-wall indications. These cracks followed the weld profile contour, and were nearly 165° in length.

The licensee stated that pre-repair ultrasonic testing (UT) examinations had identified indications in these areas during the initial inspections, but these indications had been misinterpreted as craze cracking with unusual characteristics. The characterization for these two nozzle indications was revised after the initial post-repair PT examinations. The licensee concluded that the root cause for the CRDM penetration nozzle cracking was primary water stress corrosion cracking (PWSCC). This conclusion was based on metallurgical examinations, crack location and orientation, and finite element analyses.

Discussion

The 69 CRDM nozzles at ONS3 are approximately 5 feet long and are J-groove welded to the inner radius of the RPV head, with the lower end of each nozzle extending about 6 inches below the inside of the RPV head (see Figure 2). The nozzles are constructed from 4-inch OD Alloy 600 Inconel procured in accordance with the requirements of Specification SB-167, Section II to the 1965 Edition (including Addenda through Summer 1967) of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. During initial construction, each nozzle was machined to final dimensions to assure a match between the RPV head bore and the OD of the nozzle. The nozzles were shrink-fit by cooling to at least minus 140 degrees F, inserted into the closure head penetration, and then allowed to warm to room temperature (70 degrees F minimum). The CRDM nozzles were tack-welded and then permanently welded to the closure head using 182-weld metal (see Figure 2). The shielded manual metal arc welding process was used for both the tack weld and the J-groove weld. During weld buildup, the weld was ground and PT inspected at each 9/32 inch of the weld. The final weld surface was ground and PT inspected. The weld prep for installation of each nozzle in the RPV head was accomplished by machining and buttering the J-groove with 182-weld metal.

Axial cracking in pressurized water reactor (PWR) CRDM nozzles has been previously identified, evaluated, and repaired. Numerous small-bore Alloy 600 nozzles and pressurizer heater sleeves have experienced leaks attributed to PWSCC. Generally, these components are exposed to temperatures of 600 degrees F or higher and to primary water, as are the ONS3 CRDM nozzles. However, circumferential cracks above the weld from the OD to the inside diameter (ID) have not been previously identified in the U.S.

An action plan was implemented by the NRC staff in 1991 to address PWSCC of Alloy 600 vessel head penetrations (VHPs) at all U.S. PWRs. This action plan included a review of the safety assessments by the PWR owners groups (Westinghouse Owners Group, Combustion

Engineering Owners Group, and Babcock & Wilcox Owners Group) submitted for staff review on June 16, 1993, by the Nuclear Management and Resource Council (NUMARC, now the Nuclear Energy Institute [NEI]).

After reviewing the industry's safety assessments and examining the overseas inspection findings, the NRC staff concluded, in a safety evaluation (SE) dated November 19, 1993, that PWR CRDM nozzle and weld cracking was not an immediate safety concern. The bases for this conclusion were that if PWSCC occurred (1) the cracks would be predominately axial in orientation, (2) the axial cracks would result in detectable leakage before catastrophic failure, and (3) the leakage would be detected during visual examinations performed as part of surveillance walkdown inspections before significant damage to the RPV head would occur. However, the NRC staff noted concerns about potential circumferential cracking (which would need to be addressed on a plant-specific basis), high residual stresses from initial manufacture and from tube straightening sometimes done after welding, and the need for enhanced leakage monitoring.

By letter dated March 5, 1996, NEI submitted a white paper entitled "Alloy 600 RPV Head Penetration Primary Stress Corrosion Cracking," which reviewed the significance of PWSCC in PWR VHPs, described how the PWR licensees were managing the issue. NEI assumed that the issue was primarily an economic issue rather than a safety issue, and described an economic decision tool to be used by PWR licensees to evaluate the probability of a VHP developing a crack or a through-wall leak during a plant's lifetime. This information would then be used by a PWR licensee to evaluate the need to conduct a VHP inspection at their plant.

To verify the conclusions in the industry’s safety assessments, sampling inspections were performed at three PWR units in 1994. The results of these domestic inspections were consistent with the February 1993 analyses by the PWR owners groups, the staff’s November 19, 1993, SE, and the PWSCC found in European reactors. On the basis of the results of the first five inspections of U.S. PWRs, the PWR owners groups' analyses, and the European experience, the NRC staff determined that it was probable that CRDM penetrations at other plants contained similar axial cracks, but that such cracking did not pose an immediate- or near-term safety concern. Further, the NRC staff recognized that the scope and timing of inspections may vary for different plants, depending on their individual susceptibility to this form of degradation. In the long term, however, the staff determined that degradation of the CRDM and other RPV head penetrations is an important safety consideration because of the possibility of (1) exceeding the ASME Code safety margins if the cracks are sufficiently deep and continue to propagate during subsequent operating cycles and (2) eliminating a layer of defense in depth for plant safety.

On April 1, 1997, NRC issued Generic Letter (GL) 97-01, “Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations,” which requested addressees to inform the staff of their inspection activities related to VHPs. Based on the industry’s GL 97-01 response, which took credit for periodic inspections of the RPV head, the staff agreed that the conclusions in its November 19, 1993, SE remained valid.

The recent identification of significant circumferential cracking of two CRDM nozzles at ONS3 raises concerns about a potentially risk-significant generic condition affecting all domestic PWRs. RPV head penetrations, including CRDM nozzles, provide the function of maintaining the reactor coolant system (RCS) pressure boundary. Cracking of CRDM nozzles and welds is a degradation of the primary RCS boundary. Industry experience has shown that Alloy 600 is susceptible to stress corrosion cracking (SCC). Further, the environment in the CRDM housing annulus will likely be far more aggressive after any through-wall leakage, because potentially highly concentrated borated primary water will become oxygenated, increasing crack growth rates.

The repair activities at ONS3 were extensive. The licensee stated that all flaws would be removed entirely from both weld material and nozzle base metal and repaired prior to plant restart. The licensee plans to perform a thorough visual inspection of the Unit 2 RPV head penetrations during the next outage and is investigating the eventual replacement of the RPV heads on all three units to prevent recurrence of this event. Foreign PWRs in France and Japan have already replaced a number of their RPV heads.

The NRC held a public meeting with the Electric Power Research Institute (EPRI) Materials Reliability Project (MRP) personnel on April 12, 2001, to discuss CRDM nozzle circumferential cracking issues. During the meeting, the industry representatives said that they were developing a generic safety assessment, recommendations for revisions of near-term inspections, and long-term inspection and flaw evaluation guidelines.

The ONS3 cracking reinforces the importance of examining the upper PWR RPV head area (e.g., visual under-the-insulation examinations of the penetrations for evidence of borated water leakage or volumetric examinations of the CRDM nozzles) and of using appropriate NDE methods (e.g., UT, ET, PT, etc.) to adequately characterize cracks. Presently, licensees are not required to remove RPV head insulation to visually inspect the head penetrations; however, some licensees have recently performed expanded VT-2 examinations by using cameras to inspect between the CRDM nozzles and the insulation.

The NRC has recently developed a Web page to keep the public informed of generic activities on PWR Alloy 600 weld cracking (http://www.nrc.gov/NRC/REACTOR/MRP/index.html). The NRC will update this Web page and assess the need for further generic action as new information becomes available.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

Related Generic Communications

1. Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,” March 17, 1988

2. Generic Letter 97-01, “Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations,” April 1, 1997

3. Information Notice 90-10, “Primary Water Stress Corrosion Cracking of INCONEL 600,” February 23, 1990

4. Information Notice 96-11, ”Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations,” February 14, 1996

5. NUREG/CR-6245, ”Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking,” October 1994

/RA/
Ledyard B. Marsh, Chief
Events Assessment, Generic Communications
and Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Technical contacts:
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Attachments:
1. Figure 1: Oconee Reactor Pressure Vessel Head Map
2. Figure 2: Oconee CRDM Nozzle Penetration (Typical)
3. List of Recently Issued NRC Information Notices

Figure 1: Oconee Reactor Pressure Vessel Head Map

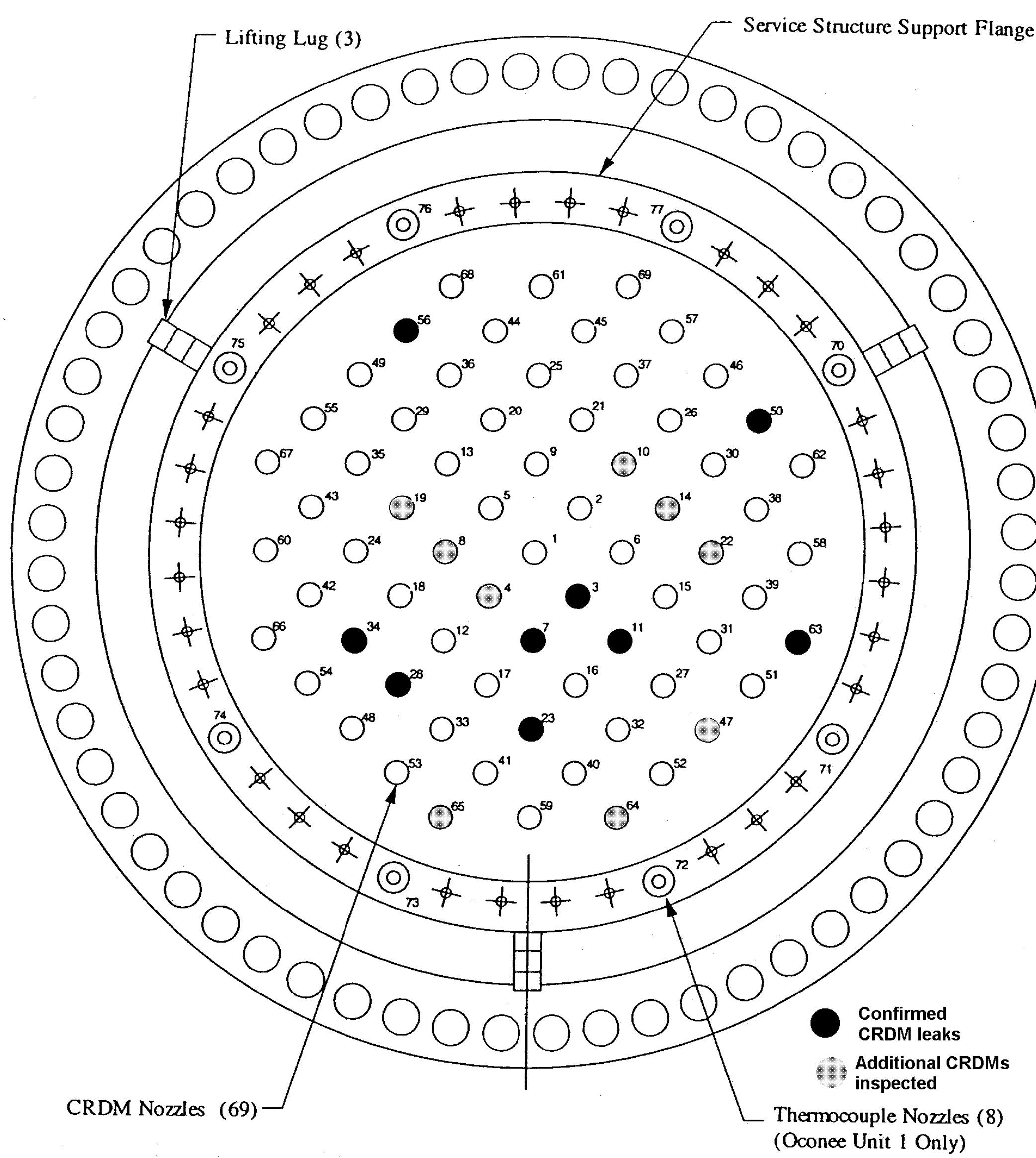
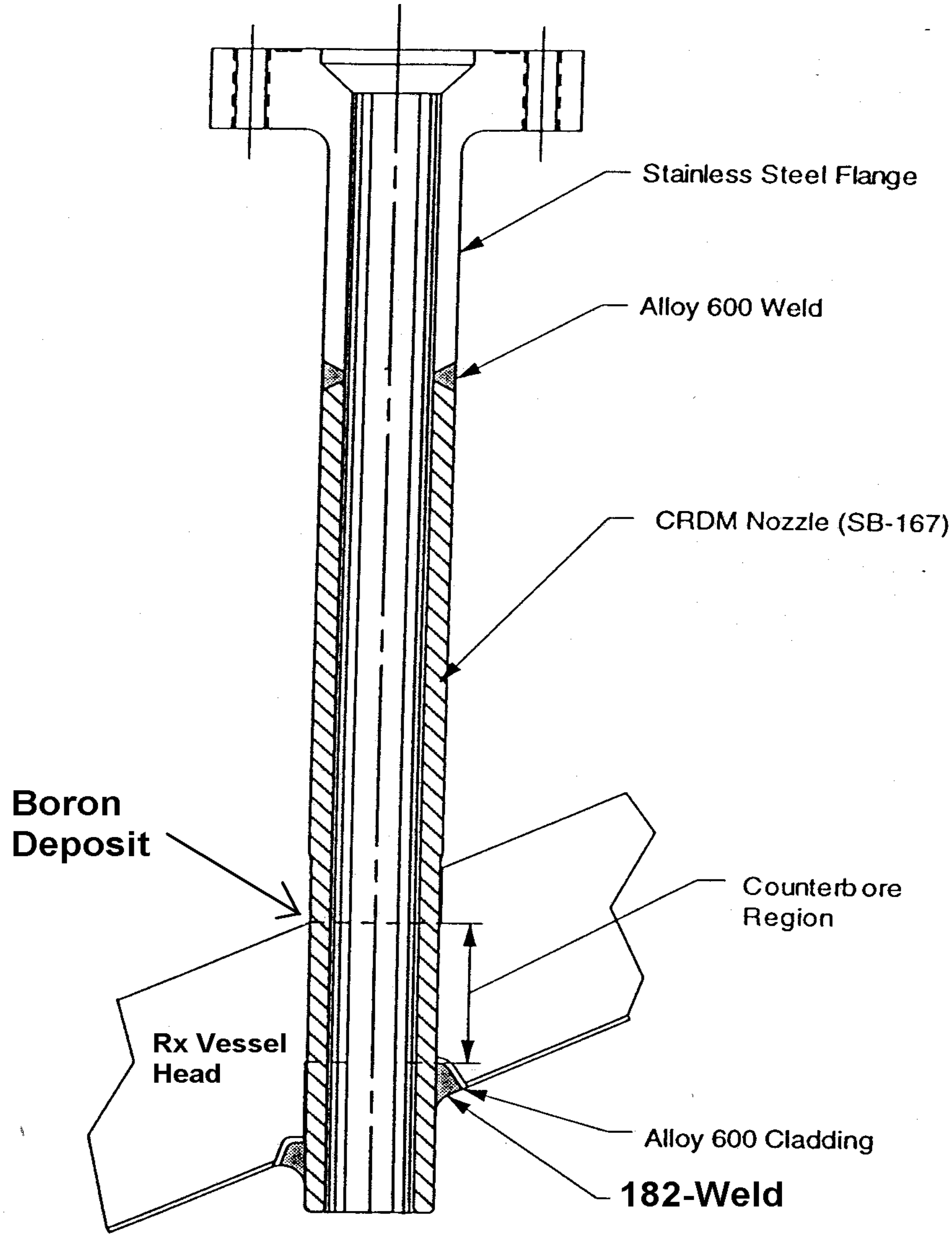


Figure 2:Oconee CRDM Nozzle Penetration (Typical)



INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	2000/06/19	

EVENT TITLE

CENTRIFUGAL CHARGING PUMP THRUST BEARING DAMAGE NOT DETECTED DUE TO INADEQUATE ASSESSMENT OF OIL ANALYSIS RESULTS AND SELECTION OF PUMP SURVEILLANCE POINTS (NRC Information Notice 2001-06)

COUNTRY	PLANT AND UNIT	REACTOR TYPE
USA	Generic	(BWR or PWR)

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

DESIGNER	1st COMMERCIAL OPERATION
(WEST, GE, CE, B&W)	N/A

ABSTRACT

This IRS report discusses inadequate assessment of pump oil analysis results that, combined with surveillance testing which does not monitor all relevant pump operating conditions, may allow severe pump degradation to go undetected.

CENTRIFUGAL CHARGING PUMP THRUST BEARING DAMAGE NOT DETECTED DUE TO INADEQUATE ASSESSMENT OF OIL ANALYSIS RESULTS AND SELECTION OF PUMP SURVEILLANCE POINTS
(NRC Information Notice 2001-06)

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

1.	**Reporting Categories:**	**1.3.3**		
2.	**Plant Status Prior to the Event:**	**2.0**		
3.	**Failed/Affected Systems:**	**3.BG**		
4.	**Failed/Affected Components:**	**4.2.7**		
5.	**Cause of the Event:**	**5.1.1.2**	**5.4.5**	
6.	**Effects on Operation:**	**6.0**		
7.	**Characteristics of the Incident:**	**7.0**		
8.	**Nature of Failure or Error:**	**8.0**		
9.	**Nature of Recovery Actions:**	**9.0**		

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D. C. 20555-0001

May 11, 2001

NRC INFORMATION NOTICE 2001-06: CENTRIFUGAL CHARGING PUMP THRUST BEARING DAMAGE NOT DETECTED DUE TO INADEQUATE ASSESSMENT OF OIL ANALYSIS RESULTS AND SELECTION OF PUMP SURVEILLANCE POINTS

Addressees

All holders of operating licenses for nuclear power reactors, 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 (NRC) is issuing this information notice to alert addressees that inadequate assessment of pump oil analysis results, combined with surveillance testing which does not monitor all relevant pump operating conditions, may allow severe pump degradation to go undetected. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid problems. However, the suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

On June 19, 2000, while disassembling the C charging/safety injection pump (CSIP) to replace a mechanical seal, Shearon Harris Nuclear Plant (SHNP) personnel discovered significant damage to the outboard thrust bearing. Further examination revealed that the babbitt material on the bearing shoes of this multi-pad thrust bearing had melted and re-solidified within the thrust bearing cage area. On both the shoes and the sleeve of the thrust bearing, radial wear in the direction of normal pump rotation was indicative of metal-to-metal contact between the two surfaces. The inboard radial bearing and shaft also had minor wear. SHNP stated in a licensee event report (Reference 1) that the most probable cause of the damage was a momentary loss of lubrication flow to the outboard thrust bearing. An inadequate fill-and-vent of the pump, which may have caused a momentary increase in the axial thrust on the outboard thrust bearing, was also given as a potential root cause.

Elemental analysis of a routine pump bearing oil sample taken on September 19, 1999, using a direct current plasma (DCP) spectrometer, revealed a 40-fold increase in the particle count in the range of 5 to 10 microns over the previous sample taken on May 11, 1999. (The particle count increased from 15,800 to 660,000 counts per 100 milliliter sample.) All other tested parameters were normal. SHNP reviewed the Electric Power Research Institute (EPRI) Lubrication Guide (Reference 2) and concluded that the size range of these wear particles was consistent with benign wear. The bearing oil in the CSIP was replaced on December 21, 1999, and SHNP continued sampling at 6-month intervals. The next oil sample, taken on February 23, 2000, also showed a high particle count in the 5 to 10 micron range. Trace amounts of iron and tin were also detected for the first time. The analysis of another oil sample taken on June 18, 2000, found that the levels of all parameters were similar to the levels in the February 23, 2000, sample.

Each CSIP at SHNP is a Pacific Model 2½ RLIJ, 11-stage, centrifugal pump manufactured by Flowserve Corporation, formerly Ingersoll-Dresser Pump Company. The C CSIP is the standby pump. During the period in which high particle counts in the three oil samples were detected, the C pump was intermittently in service to support plant operations. Surveillance testing, as required by the SHNP inservice testing program and the SHNP Technical Specifications, was performed on the C pump during this period. Inservice tests, including vibration measurement, were conducted during plant operation on November 13, 1999, and January 3, 2000, with the pump operating at the normal charging flow rate of approximately 90 gallons per minute (gpm). Performance data from both tests indicated the C CSIP met the established pump hydraulic and mechanical acceptance criteria in the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code), and no adverse trends were noted. On April 23, 2000, a refueling outage test to satisfy the SHNP technical specifications was performed successfully, with the pump achieving a flow rate of 609 gpm.

Subsequent to the discovery of the severely degraded outboard pump thrust bearing, discussions with the pump manufacturer revealed that at flow rates between approximately 250 and 600 gpm, the net axial thrust of each SHNP CSIP pump is in the direction of the outboard thrust bearing. Therefore, SHNP concluded that during normal plant operation and surveillance testing, the outboard thrust bearing had been either not loaded or only lightly loaded. In addition, SHNP could not assess the capability of the C CSIP to perform its function during a small-break, loss-of-coolant accident, in which the pump axial thrust would have fully loaded the outboard thrust bearing.

In response to an NRC notice of violation (Reference 3), SHNP described corrective steps either completed or in progress to address this issue. These included (1) counseling operators on consequences of improper pump fill-and-vent of the CSIP, (2) establishing oil analysis criteria for increased lubricant particle counts, (3) reinforcing expectations for disposition of abnormal indications, (4) sampling CSIP lubricating oil quarterly instead of semi-annually, (5) revising the maintenance procedure to ensure that the CSIP lubricating oil system will function as expected, and (6) implementing a design modification to install temperature and vibration proximity probes on each CSIP.

NRC Requirements and Industry Guidance and Practices on Pump Condition Monitoring

The current requirements for inservice testing of safety-related pumps are specified in Section 50.55a of Title 10 of the Code of Federal Regulations (10 CFR 50.55a), "Codes and Standards." For plants which are required to update their inservice testing (IST) programs after September 22, 2000, which is one year after the recent change to 10 CFR 50.55a (Reference 4), Subsection (b)(3) requires that safety-related pumps be tested to the 1995 Edition and the 1996 Addenda of the ASME OM Code. The Code requires that safety-related pumps be tested biennially at ±20% of their design flow rate, and every three months at specific reference points. Overall vibration measurements of each pump bearing are taken as specified by the Code. The SHNP IST program is in accordance with an earlier version of the Code, which requires pump testing to be conducted every three months at reference points of operation readily duplicated during subsequent tests. Pump hydraulic performance is assessed by comparing current performance with reference values established when the pump is known to be operating acceptably. Pump mechanical performance is assessed like hydraulic performance, unless the specified multiple of the measured overall vibration reference value exceeds the absolute vibration acceptance criterion.

Neither the Code nor the regulations require any specific pump condition monitoring activities to be performed on safety-related pumps. However, the NRC has observed during inspection activities that many US commercial nuclear power plants have some type of condition monitoring program for their rotating machinery. These programs usually include both safety-related and non-safety-related equipment. Because no regulations cover these programs, the testing performed, the examinations completed, and the acceptance criteria used for each condition monitoring activity vary widely.

The EPRI Lubrication Guide includes information on the testing and analysis of lubricants. The guide identifies particle size and wear-metal content as key properties to analyze. The guide also provides "classic" warning limits for certain measured properties. The guide does not recommend a specific warning limit for particle count. However, the guide emphasizes trending critical properties of a specific application and establishing appropriate warning limits. When these limits are exceeded and the results are verified, the guide recommends oil replacement and further study if necessary.

The NRC has authorized alternatives to the Code vibration requirements based on the performance of pump condition monitoring activities. For example, as part of an alternative to the Code vibration acceptance criterion, one facility committed to implement a plant-specific pump condition monitoring program for certain safety-related pumps. The NRC has determined that this proposed alternative demonstrates an acceptable level of quality and safety.

Discussion

A key factor in the failure to discover the damaged bearing before disassembly was not actively pursuing the root cause of the abnormally high particle count in the September 19, 1999, oil sample. The EPRI guide implies that particles less than 10 microns in size are generated from "benign wear." The guide does not discuss the significance of changes in wear particle concentration. However, the guide does discuss trending of parameters. SHNP performed spectroscopic analysis of each sample and trended the results of these tests. The low weight percent of the wear particles was apparently the reason why the elemental analysis did not detect the presence of bearing material. Ferrography and electron microscopic scan examination were conducted after the discovery of the bearing degradation and therefore were not a factor in diagnosing the elevated particle count. SHNP elected to continue with a routine oil sampling schedule despite the high particle count and the lack of a plausible root cause for this condition. A more aggressive oil sampling schedule (e.g., weekly) would likely have revealed the severely degraded outboard thrust bearing several months before the pump was disassembled.

Inservice and technical specification surveillance testing did not indicate that the outboard thrust bearing was severely damaged. The purpose of pump inservice testing is to identify degradation before the pump's performance of its safety-related function is impaired. For the charging pumps at the SHNP, the purpose of technical specification testing is to verify that the pump will deliver a specific flow at the required total developed head. The failure of both tests to indicate bearing degradation appears to have biased the decision to not investigate the elevated particle count.

Information provided by the vendor revealed a reversal in the direction of the pump axial force as a function of the pump flow rate. This pump design characteristic was unknown to SHNP personnel before they discovered the severely degraded bearing and then talked with the vendor. The Code does not require SHNP to account for this design condition through testing. The technical specification full flow test after the first detection of the high particle count neither detected this condition nor caused a catastrophic failure of the pump. This issue illustrates that the assessment of safety-related pump performance is dependent not only on verifying successful surveillance testing, but also on understanding (1) pump and system design and performance characteristics, (2) performance testing results, and (3) the results of condition monitoring activities and their correlation with known pump design characteristics and performance test results.

Generic Implications

If trends of condition monitoring data are not actively investigated when they deviate from an established baseline, a licensee may overlook significant pump degradation that is not detected by performance testing.

This information notice requires no specific action or written response. If you have any questions about this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

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Attachments:
1. List of References
2. List of Recently Issued Information Notices

References

1. Shearon Harris Nuclear Power Plant Unit 1, Docket Number 50-400, Licensee Event Report 2000-007-01, "Technical Specifications Violation Due to Inoperable Charging Safety Injection Pump," dated March 12, 2001.

2. NP-4916-R2, Electric Power Research Institute/Nuclear Maintenance Applications Center Lubrication Guide, Revision 2, published February 1995.

3. Shearon Harris Nuclear Power Plant Unit 1, Docket Number 05000-400, Reply to Notice of Violation (NRC Inspection Report Numbers 50-400/00-03, 50-400/00-10) dated March 2, 2001.

4. Federal Register, Volume 64, Number 183, "Industry Codes and Standards; Amended Requirements," (10 CFR Part 50), issued September 22, 1999.