December 13, 2000

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:

- LOSS OF REACTOR COOLANT INVENTORY AND POTENTIAL LOSS OF EMERGENCY MITIGATION FUNCTIONS WHILE IN A SHUTDOWN CONDITION (NRC Information Notice 95-03, Supplement 2).

- POTENTIAL DEGRADATION OF FIREFIGHTER PRIMARY PROTECTIVE GARMENTS (NRC Information Notice 2000-12).

- NON-VITAL BUS FAULT LEADS TO FIRE AND LOSS OF OFFSITE POWER (NRC Information Notice 2000-14).

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 Eric J. Benner of my staff. He can be reached at (301) 415-1171.

Sincerely,

/RA John R. Tappert For/
Ledyard B. Marsh, Chief
Events Assessment, Generic Communications 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
Regulatory Activities Unit
Safety Assessment Section
Division of Nuclear Installation Safety
International Atomic Energy Agency
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P.O. Box 100, A-1400
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Le Seine Saint Germain
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92130, Issy-les-Moulineaux, France

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INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	2000/07/03	

EVENT TITLE

LOSS OF REACTOR COOLANT INVENTORY AND POTENTIAL LOSS OF EMERGENCY MITIGATION FUNCTIONS WHILE IN A SHUTDOWN CONDITION
(NRC Information Notice 95-03, Supplement 2)

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 insights from inspections related to losses of reactor coolant while the reactor is in a hot, pressurized, shutdown condition with the potential for adversely affecting accident mitigation capability. At one facility, the procedure used to recover from a hot, pressurized shutdown LOCA directs operator actions that could increase the probability that ECCS equipment would be disabled and could delay recovery from the LOCA. Specifically, for entry conditions in which the pressurizer level is less than 10 percent or RCS subcooling is less than 32 degrees C (58 degrees F), and the RWST level is greater than 27 percent, closure of the RHR hot-leg isolation valves are not explicitly directed early in the procedure, and, as such, the LOCA is not terminated. The procedure subsequently directs alignment of the suction of the operating centrifugal charging pump to the common suction header, potentially exposing it to hot reactor coolant and rendering it inoperable. If that centrifugal charging pump is rendered inoperable, the operator is expected to evaluate aligning the suction of the second centrifugal charging pump to the common suction header, exposing it to the same conditions which rendered the first centrifugal charging pump inoperable. This progression continues with the two safety injection pumps, potentially rendering them inoperable, also, and could ultimately result in a significant loss of ECCS capability.

LOSS OF REACTOR COOLANT INVENTORY AND POTENTIAL LOSS OF EMERGENCY MITIGATION FUNCTIONS WHILE IN A SHUTDOWN CONDITION
(NRC Information Notice 95-03, Supplement 2)

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.BG		
4.	Failed/Affected Components:	4.2.1		
5.	Cause of the Event:	5.5.7		
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

July 3, 2000

NRC INFORMATION NOTICE 95-03, SUPPLEMENT 2: LOSS OF REACTOR COOLANT INVENTORY AND POTENTIAL LOSS OF EMERGENCY MITIGATION FUNCTIONS WHILE IN A SHUTDOWN CONDITION

Addressees

All holders of operating licenses for 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 supplement to alert addressees to insights from inspections related to losses of reactor coolant while the reactor is in a hot, pressurized, shutdown condition with the potential for adversely affecting accident mitigation capability. It is expected that recipients will review the information in this supplement for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice supplement are not NRC requirements; therefore, no specific action or written response is required.

Background

Information Notice 95-03, "Loss of Reactor Coolant Inventory and Potential Loss of Emergency Mitigation Functions While in a Shutdown Condition," issued on January 18, 1995, discusses an incident at Wolf Creek involving the loss of reactor coolant while the reactor was in a hot, pressurized, shutdown condition. At Wolf Creek, all residual heat removal (RHR) and emergency core cooling system (ECCS) pump suction lines are supplied by a common suction header. When the loss of reactor coolant occurred, hot reactor coolant system (RCS) water was introduced into this common suction header between the refueling water storage tank (RWST) and the RHR and ECCS pumps. This hot water flashed to steam, resulting in a steam/water mixture in the header. Had an ECCS actuation occurred, this mixture could have been introduced into the suction of the ECCS pumps. If operators had not been able to terminate the event, the hot water in the suction piping may have led to steam binding, which could have adversely affected the pumps in both ECCS trains. In addition, water flashing to steam in the header and the RWST could have caused serious mechanical damage to the RHR piping and the RWST as a result of water hammer. Finally, the flow path from the RCS through the RWST establishes a containment bypass path.

ML003727420

The NRC issued Generic Letter (GL) 98-02, "Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition," on May 28, 1998, to request that addressees (1) perform an assessment to determine if their ECCSs include certain design features, such as a common pump suction header, which can render the systems susceptible to common-cause failure and if this susceptibility is found, (2) prepare, with consideration of plant-specific design attributes, a description of the features of their Appendix B quality assurance program (for example, the methods used to verify valve position; the controls in place to ensure compliance with plant surveillance, maintenance, modification, and operating procedures; and the adequacy of operator training for such activities) that provide assurance that the safety-related functions of the RHR system and the ECCS will not be adversely affected by activities conducted at hot shutdown.

This information notice supplement discusses additional insights that have been gained from NRC inspections performed in response to GL 98-02.

Discussion

Licensee and staff analyses of the Wolf Creek event assumed failure to isolate the RCS draindown path. If this were to occur, within 5 minutes, the reactor vessel water level could drain to the bottom of the hot leg, and, as a consequence, the operating RHR pump would lose suction, cavitate, and fail. Continued boiloff would result in uncovering of the core in less than 1 hour. Failure to quickly isolate the flow path could also result in the RWST suction header filling with steam, which would continually discharge into the RWST. This steam could lead to water hammer events with the potential for mechanical damage to pump components, piping, and tank structural components.

A special NRC report dated March 1995 and titled "Reactor Coolant System Blowdown at Wolf Creek on September 17, 1994" (AEOD/S95-01), identified 19 loss-of-coolant events that have occurred at reactors during shutdown. Of these 19 events, only 2 have taken place at temperatures and pressures sufficient to result in voiding RWST piping. Considering pressurized-water reactor operating experience, the staff estimated the initiating event frequency may be equal to or greater than 1E-3 per reactor year. The initiating event frequency and the heavy dependence upon short-term operator action highlight the importance of careful planning, the accuracy of administrative procedures, and disciplined adherence to those procedures.

GL 98-02 focused on requesting that affected licensees take adequate protective measures to minimize the likelihood of a Wolf Creek type event. The generic letter was not intended to address all aspects related to recovery from a shutdown loss-of-coolant accident (LOCA). However, during NRC inspections performed in response to GL 98-02, the staff identified a potentially generic weakness regarding the procedures used to mitigate a Wolf Creek type event that licensees may wish to review.

At one facility, the procedure used to recover from a hot, pressurized shutdown LOCA directs operator actions that could increase the probability that ECCS equipment would be disabled and could delay recovery from the LOCA. Specifically, for entry conditions in which the pressurizer level is less than 10 percent or RCS subcooling is less than 32 degrees C (58 degrees F), and the RWST level is greater than 27 percent, that is, a Wolf Creek type scenario,

closure of the RHR hot-leg isolation valves are not explicitly directed early in the procedure, and, as such, the LOCA is not terminated. The procedure subsequently directs alignment of the suction of the operating centrifugal charging pump to the common suction header, potentially exposing it to hot reactor coolant and rendering it inoperable. If that centrifugal charging pump is rendered inoperable, the operator is expected to evaluate aligning the suction of the second centrifugal charging pump to the common suction header, exposing it to the same conditions which rendered the first centrifugal charging pump inoperable. This progression continues with the two safety injection pumps, potentially rendering them inoperable, also, and could ultimately result in a significant loss of ECCS capability.

Related Generic Communications

- GL 98-02, "Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition," May 28, 1998.
- Information Notice 95-03, "Loss of Coolant Inventory and Potential Loss of Emergency Mitigation Functions While in a Shutdown Condition," January 18, 1995.
- Supplement to Information Notice 95-03, March 25, 1996.

This information notice requires no specific action or written response. However, recipients are reminded that they are required to consider industry-wide operating experience (including NRC information notices), where practical, when setting goals and performing periodic evaluations under Section 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," of Part 50 of Title 10 of the Code of Federal Regulations. If you have any questions about the information in this notice, please contact the technical contact listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

Original signed by

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

Technical contact: Eric J. Benner, NRR
301-415-1171
E-mail: ejb1@nrc.gov

Attachment: List of Recently Issued NRC Information Notices

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	2000/09/21	

EVENT TITLE

POTENTIAL DEGRADATION OF FIREFIGHTER PRIMARY PROTECTIVE GARMENTS
(NRC Information Notice 2000-12)

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 potential degradation of performance of firefighter primary protective garments (FFPPGs). The licensee for Hope Creek Nuclear Power Plant reported that following a fire event at the plant it outsourced its FFPPGs to a firm that specializes in decontamination and repair of FFPPGs. The firm found that some of the FFPPGs were degraded and had to be condemned while others had to be repaired. None of this damage was attributed to the fire event; rather, it was all attributed to storage practices and normal wear.

POTENTIAL DEGRADATION OF FIREFIGHTER PRIMARY PROTECTIVE GARMENTS
(NRC Information Notice 2000-12)

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.KH		
4.	Failed/Affected Components:	4.0		
5.	Cause of the Event:	5.4.23		
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

September 21, 2000

NRC INFORMATION NOTICE 2000-12: POTENTIAL DEGRADATION OF FIREFIGHTER PRIMARY PROTECTIVE GARMENTS

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 potential degradation of performance of firefighter primary protective garments (FFPPGs). It is expected that 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

The licensee for Hope Creek Nuclear Power Plant reported that following a fire event at the plant it outsourced its FFPPGs to a firm that specializes in decontamination and repair of FFPPGs. The firm found that some of the FFPPGs were degraded and had to be condemned while others had to be repaired. None of this damage was attributed to the fire event; rather, it was all attributed to storage practices and normal wear. Informal NRC staff discussions with representatives at several other nuclear power plants indicated that licensees may not be aware of this problem.

Discussion

Moisture Barrier Degradation

Aldan Engineered Coated Fabrics indicated in a letter dated November 3, 1999, http://www.usfa.fema.gov/alerts/breathetex.htm, that its product BREATHE-TEX has shown signs of degradation under certain conditions. BREATHE-TEX is a trade name for a material that is used as a moisture barrier in FFPPGs. In a letter from Edwin T. Winter, Chairman and CEO of Aldan Engineered Coated Fabrics, to manufacturers of FFPPGs, the degradation is described as follows:

> "The degradation, not readily observable by routine visual inspection, is primarily in the form of film cracking. A degraded moisture barrier film will allow the passage of liquids, thereby reducing the level of protection in proportion to the degree of degradation."

ML003740412

In a joint statement issued by manufacturers of FFPPGs, dated November 15, 1999, http://www.turnout-info.com/en/news/joint_stmt.htm (note there is an underline "_" between "joint" and "stmt" in this web address) they remind users of FFPPGs that the moisture barriers are the "most fragile component in your gear [FFPPGs]" and that FFPPGs should be inspected on a regular basis. They also state that "particular attention should be paid to the moisture barrier, regardless of what brand it is."

Outer Shell Degradation

Lawrence Livermore National Laboratory (LLNL) report titled "Statistics for the Time-Dependent Failure of Kevlar-49/Epoxy Composites: Micromechanical Modeling and Data Interpretation" by S. L. Phoenix and E. M. Wu indicates that Kevlar, a common fabric used in the production of the durable and heat-resistant outer shell of FFPPGs, is susceptible to photodegradation from ultraviolet (UV) light sources.

Photodegradation is a phenomenon in which the tensile strength of the fibers is reduced as a result of exposure to UV light sources such as sunlight and fluorescent light. Photodegradation leads to reduced abrasion and tear resistance in aramid fibers such as Kevlar.

Many fabrics used in FFPPGs (see list quoted below) use aramid fibers that are subject to photodegradation. Therefore, special care must be taken to ensure that the fabrics are protected from UV light sources. The User Instruction, Safety and Training Guide provided by Lion Apparel (Dayton, Ohio) gives the following warning:

> "Exposure to ultraviolet light (found in the sun's rays and fluorescent light) will severely weaken and damage the fabrics in your protective clothing after only A FEW DAYS. This is especially true for fabrics of the following aramid materials: Hoechst Celanese Pbi, Dupont Kevlar, Dupont Nomex, Dupont Nomex Omega, Dupont Nomex IIIA, Lenzing P84, Southern Mills Advance, and BASF Basofil."

FFPPGs should be dried after cleaning in the shade rather than in a place exposed to UV light.

Users of FFPPGs should consider photodegradation as a damage mechanism. This degradation may not be readily apparent but may seriously degrade the protection offered to plant firefighters. Regular inspection and testing should be performed in accordance with the recommendation of manufacturers and distributers.

Conclusion

Proper storage and regular thorough inspection of FFPPGs are important to ensure that FFPPGs provide effective firefighter protection. FFPPG degradation may not be readily apparent; therefore, special training may be required for inspection of FFPPGs.

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

/RA/

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

/RA/

Ledyard B. Marsh, Chief
Events Assessment, Generic Communications
and Non-Power Reactors Branch
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Office of Nuclear Reactor Regulation

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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
	2000/09/27	

EVENT TITLE

NON-VITAL BUS FAULT LEADS TO FIRE AND LOSS OF OFFSITE POWER
(NRC Information Notice 2000-14)

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 equipment and design issues identified following a recent transient at the Diablo Canyon nuclear power plant. The aspect of the transient considered noteworthy was the failure of bus duct, a passive component of known high reliability which often receives little preventive maintenance or attention. On May 15, 2000, at Diablo Canyon Unit 1, a phase-to-phase electrical fault occurred in a 12-kV non-Class 1E electrical bus duct from the unit auxiliary transformer to the switchboards that supplied the reactor coolant pumps and the circulating water pumps. The fault caused a turbine trip and consequent reactor trip. As this section of bus could not be isolated from the main generator, the fault lasted for 4 to 8 seconds until the main generator electrical field voltage decayed. The 12-kV bus fault occurred at a point at which the bus duct passes under the 4-kV non-Class 1E bus from the startup transformer. The original fault and the resultant arcing and smoke caused another fault, this time in the 4-kV bus duct directly above the original failure.

NON-VITAL BUS FAULT LEADS TO FIRE AND LOSS OF OFFSITE POWER
(NRC Information Notice 2000-14)

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:	2.1.1		
3.	Failed/Affected Systems:	3.EB	3.EC	3.EE
4.	Failed/Affected Components:	4.3.1		
5.	Cause of the Event:	5.1.2.2		
6.	Effects on Operation:	6.1		
7.	Characteristics of the Incident:	7.9		
8.	Nature of Failure or Error:	8.2		
9.	Nature of Recovery Actions:	9.2		

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

September 27, 2000

NRC INFORMATION NOTICE 2000-14: NON-VITAL BUS FAULT LEADS TO FIRE AND LOSS OF OFFSITE POWER

Addressees

All holders of licenses for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees of equipment and design issues identified following a recent transient at the Diablo Canyon nuclear power plant. The aspect of the transient considered noteworthy was the failure of bus duct, a passive component of known high reliability which often receives little preventive maintenance or attention. It is expected that 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 actions or written response is required.

Description of Circumstances

On May 15, 2000, at Diablo Canyon Unit 1, a phase-to-phase electrical fault occurred in a 12-kV non-Class 1E electrical bus duct from the unit auxiliary transformer to the switchboards that supplied the reactor coolant pumps and the circulating water pumps. The fault caused a turbine trip and consequent reactor trip. As this section of bus could not be isolated from the main generator, the fault lasted for 4 to 8 seconds until the main generator electrical field voltage decayed. The 12-kV bus fault occurred at a point at which the bus duct passes under the 4-kV non-Class 1E bus from the startup transformer. The original fault and the resultant arcing and smoke caused another fault, this time in the 4-kV bus duct directly above the original failure.

The 12-kV circuit breaker that supplied the 4-kV startup transformers and the faulted 4-kV bus duct downstream of the transformer tripped in response to this second fault. This trip resulted in a loss of power to all 4-kV vital (safety-related) and non-vital buses. All three diesel generators started and all vital loads were re-energized. However, the combination of the two faults disabled both power supplies to the non-vital 4-kV buses.

Besides de-energizing all non-vital 4-kV power within the plant, the loss of both sources of non-vital power also caused a loss of the 480-Vac power supply to the switchyard control building. This failure led to a loss of power to the charger for the switchyard batteries; the eventual depletion of the switchyard batteries would have led to a loss of control power in the switchyard serving both Diablo Canyon units. The loss of control power would have disabled remote control of the switchyard high voltage circuit breakers.

ML003748744

The licensee installed a portable generator to restore power to the charger before the switchyard batteries were depleted. On May 16, 2000, after 33 hours, plant personnel energized the 4-kV and 480-Vac non-vital buses by backfeeding through Auxiliary Transformer 1-2.

Discussion

Switchgear Room Arrangement and Bus Duct Construction

The auxiliary and startup transformers are connected to the onsite distribution switchgear by bus bars with a ½ - by 6-inch cross-sectional area. All three phases are enclosed in a single aluminum duct (nonsegregated). The startup and auxiliary 12-kV and 4-kV non-vital switchgear are located within a common room. To connect the two sources of offsite power to multiple switch boards within the room, there are many crossing bus ducts above the switchgear. The non-vital 4-kV bus ducts from both auxiliary and startup power are in close proximity for extended runs. Since none of the bus ducts was designed as safety grade, no regulatory separation criteria apply.

The bus work in the room was a combination of aluminum and copper bus bars connected with aluminum splice plates secured by four ½-inch bolts. The bus bars and splice plates were silver-plated at the connection points to ensure conductivity. The bus bars had a nominal 3/16-inch gap between them at the splice plate to allow for thermal expansion. The 12-kV bus bars had a 6-inch air gap between phases, which is slightly below the required air gap for uninsulated conductors, so bus bars and connections were insulated with a combination of sleeves and wraps. The 4-kV bus bars were similarly insulated.

Root Cause

The licensee's evaluation concluded that a center bus bar overheated at a splice joint, which caused a polyvinyl chloride boot insulator over the splice joint to smoke. Eventually, heat-induced failure of fiberglass insulation on adjacent phases resulted in phase-to-phase arcing. The fault and resultant fire destroyed any direct physical evidence of the root cause; however, the factors discussed below could individually or jointly have led to the failure. They include inconsistent silverplating, currents approaching bus capacity, undersized splice plates, torque relaxation of connecting bolts, and undetected damage from a 1995 explosion of Auxiliary Transformer 1-1.

(1) Silverplating

Many of the bus bars and splice plates had only a thin layer of silverplating. Laboratory analysis determined that the silverplating on one splice plate had partially separated from the base aluminum, and corrosion products were found on the aluminum surface. If this separation had existed at the point of the fault, it would have created higher resistance and, therefore, more heat at the connection. The laboratory stated that the most likely source of corrosive compounds was the polyvinyl chloride insulating boot. Silverplating was also observed flaking off the aluminum bus bars at two other splice joints not directly affected by the fault.

(2) Heavy Bus Loading and Splice Joint Configuration

The 12-kV 6-inch aluminum bus bars were rated at 2250 amps. The bus was routinely loaded to 2100 amps with an actual worst case operating load of about 2250 amps. The vendor stated that all bus bars supplied to Diablo Canyon met the design requirements of Institute of Electrical and Electronics Engineers (IEEE) 37.20-1969, "IEEE Standard for Switchgear Assemblies Including Metal-Enclosed Bus," which stipulates an operating temperature limit of 65 C. Vendor-supplied test data reported a maximum temperature rise of 46 C at 2000 amps and 63 C at 2200 amps for the bus bar type that failed. Since the test temperature increased 17 C for a current increase of 200 amps, a temperature increase of only 2 C for an additional load of 50 amps seems improbably low and it is reasonable that the bus and its insulation had exceeded design conditions for some time. In addition, the vendor heat rise tests of aluminum bus bars for 2200 amps were conducted with two splice plates at the splice joint. The vendor test used 3 - by 4-inch copper splice plates instead of the 2½ - by 4-inch aluminum splice plates used at Diablo Canyon.

The inspectors noted that the splice plates connecting the bus bars were considerably smaller than the bus bars themselves. Some splice joints had two aluminum plates on each phase sized 2½ inches by 4 inches by ⅜ inch. Also, the splice plates were not always centered between the bus bars. The lack of centering of splice plates and splice plates smaller than the tested configuration reduced the contact area, causing increased heat generation.

(3) Torque Relaxation

The as-found torque values for many of the splice plate connecting bolts was 10 to 20 foot-pounds indicating that thermal relaxation had occurred since initial installation. The bolts had an initial torque value of 40 foot-pounds. Torque relaxation on the bolts could have allowed the splice plate to lose contact with the bus bar, leading to arcing and overheating.

(4) Undetected Damage from 1995 Auxiliary Transformer 1-1 Explosion

A 1995 explosion of Auxiliary Transformer 1-1 had displaced the 12-kV busing several feet into the turbine building. Most of the bus bar connections upstream of the failed connection had been disassembled and repaired where necessary. Records for the failed connection were incomplete. The inspectors determined that the failed joint had been visually inspected and micro-ohm tested; however, no evidence was found to verify that the joint had been torqued. Inadequate torque could have resulted in increased resistance and heat generation if the joint became loose.

Corrective Actions

All damaged components were repaired and refurbished, accessible splice joints on the 12-kV auxiliary bus and 4-kV startup bus were inspected and torqued, and post-maintenance tests were conducted to ensure that the bus bars were properly restored. The licensee concluded that a similar defect was unlikely to occur on Unit 2 because the splice plates for the Unit 2 bus, although similar in construction to the one that failed in Unit 1, had previously been inspected

and torqued. The licensee also tested the affected startup and auxiliary transformers to ensure no breakdown had occurred in the winding insulation.

The licensee examined the various design issues associated with the 12-kV and 4-kV buses. The buses with operating currents near design limits were judged to be operable because the expected temperature at the Diablo Canyon site was considerably lower than the ambient temperature assumed in the vendor heat rise test acceptance criteria. The licensee grouped renovation and maintenance of the undamaged runs of bus into three groups by decreasing order of failure susceptibility to prioritize remedial actions: (1) bus sections where normal load has little margin relative to the continuous duty design rating, (2) bus sections in which the auxiliary buses pass near the startup buses, and (3) bus sections that cannot be isolated from the main generator (i.e., have no breaker to quickly sense and interrupt the fault).

During upcoming refueling outages, planned actions to prevent recurrence include the following: (1) inspect and torque the booted connections on the 3750-amp bus for Startup Transformer 1-1 and replace splice plates on the booted connections with full-face copper splice plates; (2) inspect and torque splice plate connections on the 4-kV auxiliary buses; and (3) upgrade the 2250-amp 12-kV buses from aluminum to copper bars. The same inspection and replacement activities will occur on taped connections during the subsequent Unit 1 refueling outage.

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.

/RA/Marvin M. Mendonca FOR

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