
Appendix E: Generic Communication and Compliance Activities

This appendix documents those generic communication and compliance activities (GCCA) completed by NRR that did not meet the criteria for designation as generic issues (GI), but were important enough to require the issuance of Information Notices (IN) and/or Generic Letters (GL) to licensees. The plan for documenting closed

GCCAs was delineated in SECY-96-107.¹ On this page:

- *GCCA-0001: ASSESSMENT OF CONDITION OF SAFETY-RELATED STRUCTURES AND CIVIL ENGINEERING FEATURES*
- *GCCA-0002: ENVIRONMENTAL LICENSING AND REGULATORY CONCENTRATIONS IN BUILDING WAKES*
- *GCCA-0003: RRG, 50.54(P) GUIDANCE*
- *GCCA-0004: RELOCATION OF SELECTED TS REQUIREMENTS RELATED TO INSTRUMENTATION (GL)*
- *GCCA-0005: BWR - SCRAM SOLENOID PILOT VALVE PROBLEMS*
- *GCCA-0006: SHIFT STAFFING ISSUE FOLLOWUP*
- *GCCA-0007: LESSONS LEARNED FROM OPERATIONAL SAFEGUARDS RESPONSE EVALUATIONS*
- *GCCA-0008: AIR ENTRAINMENT IN TERRY TURBINE LUBRICATING CONTROL OIL SYSTEM*
- *GCCA-0009: WRONG REPLACEMENT PARTS RELIEF VALVES AND REFUELING MAST*
- *GCCA-0010: SPENT FUEL POOL OVERFLOW INTO VENTILATION SYSTEM*
- *GCCA-0011: IPEEE FOR SEVERE ACCIDENT VULNERABILITIES*
- *GCCA-0012: SURRY VENTILATION FILTER ISSUE*
- *GCCA-0013: COMMON MODE FAILURE OF COPEES VOLCAN PORVs*
- *GCCA-0014: DEFICIENCIES IDENTIFIED DURING ELECTRICAL DISTRIBUTION SYSTEM*
- *GCCA-0015: SEISMIC ADEQUACY OF THERMO-LAG PANELS*
- *GCCA-0016: CAPABILITY OF OFFSITE POWER DURING DESIGN BASIS EVENTS*
- *GCCA-0017: POTENTIAL FOR LOSS OF AUTOMATIC ESF ACTUATION*
- *GCCA-0018: POTENTIAL FOR MOV FAILURE - STEM PROTECTION PIPE CHANGES*
- *GCCA-0019: UNANTICIPATED AND UNAUTHORIZED MOVEMENT OF FUEL*
- *GCCA-0020: FRAUDULENT COMMERCIAL GRADE CERTIFICATE OF COMPLIANCE*
- *GCCA-0021: CHATTER OF ITT BARTON 288A AND 289A DIFFERENTIAL PRESSURE*
- *GCCA-0022: FREQUENCY OF USE OF AIR-OPERATED GATE VALVES*
- *GCCA-0023: PRESSURE LOCKING AND THERMAL BINDING OF GATE VALVES*
- *GCCA-0024: DEGRADED DECAY HEAT REMOVAL CAPABILITY VIA NATURAL CIRCULATION*
- *GCCA-0025: FALSIFICATION OF ASNT CERTIFICATE BY AMERICAN POWER SERVICES*
- *GCCA-0026: ADEQUACY OF EMERGENCY AND ESSENTIAL LIGHTING*

¹ SECY-96-107, "Uniform Tracking of Agency Generic Technical Issues," U.S. Nuclear Regulatory Commission, May 14, 1996. [9605230140]

- *GCCA-0027: ADDRESS CONCERNS REGARDING ASME CODE*
- *GCCA-0028: CIRCUMFERENTIAL CRACKING OF STEAM GENERATOR TUBES*
- *GCCA-0029: REACTOR COOLANT PUMP TURNING VANE BOLT LOCKING DEVICE FAILURE*
- *GCCA-0031: POTENTIAL CABLE DAMAGE FROM EXCESS SIDE WALL PRESSURE*
- *GCCA-0032: EVALUATE MISSILES FROM MIRROR INSULATION DURING HIGH ENERGY PIPE BREAKS*
- *GCCA-0033: RESULTS OF RECENT NRC SPONSORED FLAME SPREAD AND FIRE ENDURANCE TESTING*
- *GCCA-0036: FAILURE OF AUTOMATIC VENTILATION SYSTEM OPERATION FOLLOWING A LOSS OF OFFSITE POWER*
- *GCCA-0037: FAILURE TO TEST SWING BUSES DURING INTEGRATED EMERGENCY DIESEL GENERATOR SURVEILLANCE*
- *GCCA-0038: LIGHTNING DISSIPATION SYSTEMS*
- *GCCA-0039: SWITCHGEAR FIRE AND PARTIAL LOSS OF OFFSITE POWER*
- *GCCA-0040: SPENT FUEL TRANSFER CANAL SHIELDING DEFICIENCY AT BOILING WATER REACTOR*

- *GCCA-0041: CHANGES IN THE OPERATOR LICENSING PROGRAM*
- *GCCA-0042: UNPLANNED, UNMONITORED RELEASE OF RADIOACTIVITY FROM THE EXHAUST VENTILATION SYSTEM OF A BWR*
- *GCCA-0043: SUSCEPTIBILITY OF LOW PRESSURE COOLANT AND CORE SPRAY INJECTION VALVES TO PRESSURE LOCKING*
- *GCCA-0044: POTENTIALLY NONCONFORMING FASTENERS SUPPLIED BY A&G ENGINEERING II, INC.*
- *GCCA-0045: CURRENT FIRE ENDURANCE TEST RESULTS FOR 3M INTERAM RACEWAY FIRE BARRIER SYSTEMS*
- *GCCA-0046: POTENTIAL FOR DATA COLLECTION EQUIPMENT TO AFFECT PROTECTION SYSTEM PERFORMANCE*
- *GCCA-0047: BORAFLEX DEGRADATION IN SPENT FUEL POOL STORAGE RACKS*
- *GCCA-0049: LEGAL ACTIONS AGAINST THERMAL SCIENCE, INC., MANUFACTURER OF THERMO-LAG*
- *GCCA-0050: TRANSIENT INVOLVING OPEN SAFETY RELIEF VALVE FOLLOWED BY COMPLICATIONS*
- *GCCA-0052: UNEXPECTED OPENING OF AN SRV AND COMPLICATIONS INVOLVING SUPPRESSION POOL STRAINER BLOCKAGE*
- *GCCA-0053: DECAY HEAT MANAGEMENT PRACTICES DURING REFUELING*
- *GCCA-0054: POTENTIAL FOR LOSS OF AUTOMATIC ENGINEERED SAFETY FEATURES ACTUATION*
- *GCCA-0056: AUGMENTED REACTOR VESSEL INSPECTION*
- *GCCA-0057: CONSIDERATION OF POSITION CHANGEABLE VALVES*
- *GCCA-0058: PROBLEM OF GREASE LEAKAGE IN PRE-STRESSED CONCRETE CONTAINMENT*
- *GCCA-0060: INADEQUATE TESTING OF SAFETY-RELATED LOGIC CIRCUITS*
- *GCCA-0061: BORAFLEX DEGRADATION IN SPENT FUEL POOL STORAGE RACKS*
- *GCCA-0062: ANSYS AND GTSTRUDL COMPUTER PROGRAM ERROR NOTIFICATIONS*
- *GCCA-0063: INADEQUATE CONTROL OF MOLDED-CASE CIRCUIT BREAKERS*
- *GCCA-0064: RELOCATION OF RCS PRESSURE/TEMPERATURE LIMITS*
- *GCCA-0065: RECONSIDERATION OF PLANT SECURITY REQUIREMENTS*
- *GCCA-0066: FIRES IN EMERGENCY DIESEL GENERATOR EXCITERS*
- *GCCA-0067: INADEQUATE CAPACITY OF CCW LEADS TO FREON RELEASE TO THE CONTROL ROOM*
- *GCCA-0068: BWR STABILITY WITH FLOW SLIGHTLY LESS THAN NATURAL CIRCULATION FLOW*
- *GCCA-0070: EVALUATE IMPACT OF RCP SUPPORT COLUMN TILT ON LEAK BEFORE BREAK ANALYSES*
- *GCCA-0071: FISH MOUTH BURST AND BOWING OF PREVIOUSLY-PLUGGED STEAM GENERATOR TUBES*
- *GCCA-0072: BLOCKAGE OF UNTESTED ECCS PIPING*

- *GCCA-0073: PORV INOPERABILITY MASKED BY DOWNSTREAM INDICATIONS DURING TESTING*
- *GCCA-0074: LOSS OF RC INVENTORY AND POTENTIAL LOSS OF EMERGENCY MITIGATION FUNCTIONS WHILE IN A SHUTDOWN CONDITION*
- *GCCA-0075: CONTROL ROD DRIVE MECHANISM PENETRATION CRACKING*
- *GCCA-0076: AUGMENTED EXAMINATION OF REACTOR VESSEL*
- *GCCA-0077: CLOSED HEAD VENT CAUSES INACCURATE LEVEL INDICATION DURING REDUCED INVENTORY*
- *GCCA-0078: SHUTDOWN COOLING FLOW BYPASSING CORE RESULTS IN TEMPERATURE AND PRESSURE INCREASES*
- *GCCA-0079: POTENTIAL CONTAINMENT LEAK PATH THROUGH HYDROGEN ANALYZER*
- *GCCA-0080: INADEQUATE TESTING AND DESIGN OF TORNADO DAMPERS*
- *GCCA-0081: ASSESSMENT OF CORROSION OF B&W FUEL USED IN 2-YEAR FUEL CYCLES*
- *GCCA-0082: ENVIRONMENTAL EFFECTS ON MAIN STEAM SAFETY VALVE SET POINT*
- *GCCA-0083: INADVERTENT DRAINING OF REACTOR VESSEL AND ISOLATION OF SHUTDOWN COOLING SYSTEM*
- *GCCA-0084: RECENT PROBLEMS WITH OVERHEAD CRANES*

- *GCCA-0085: REMOVING REFUELING FLOOR SHIELDING PLUGS PRIOR TO AND SOON AFTER SHUTDOWN*
- *GCCA-0086: DAMAGE TO VALVE INTERNALS CAUSED BY THERMALLY-INDUCED PRESSURE LOCKING*
- *GCCA-0087: DAMAGE IN FOREIGN STEAM GENERATOR INTERNALS*
- *GCCA-0088: INTERFACE BETWEEN OPERATORS AND NUCLEAR ENGINEERS DURING TESTS AND STARTUP*
- *GCCA-0089: VALVE STEM COUPLING OF GIMPEL AUXILIARY FEEDWATER TURBINE TRIP THROTTLE VALVES*
- *GCCA-0091: USE OF INDIVIDUAL PLANT EXAMINATIONS (IPEs) FOR REGULATORY DECISION MAKING*
- *GCCA-0092: OVERWITHDRAWAL OF TIP*
- *GCCA-0093: SPENT FUEL POOL COOLING*
- *GCCA-0094: SOUTH TEXAS STUCK ROD EVENT FOLLOWING REACTOR TRIP*
- *GCCA-0095: RADWASTE FACILITY EQUIPMENT DEGRADATION AT MILLSTONE UNIT 1*
- *GCCA-0096: WOLF CREEK REACTOR TRIP WITH ONE TRAIN ESSENTIAL SERVICE WATER SYSTEM INOPERABLE*
- *GCCA-0097: STUCK CONTROL ROD PROBLEMS*
- *GCCA-0099: SLOW FIVE PERCENT SCRAM INSERTION TIMES CAUSED BY VITON DIAPHRAGMS IN SCRAM SOLENOID PILOT VALVES*
- *GCCA-0100: POTENTIAL CLOGGING OF HPSI THROTTLE VALVES DURING CONTAINMENT SUMP RECIRCULATION PHASE*
- *GCCA-0101: STEAM GENERATOR TUBE INSPECTION RESULTS*
- *GCCA-0102: REACTOR OPERATION BELIEVED TO BE INCONSISTENT WITH THAT DESCRIBED IN THE FSAR*
- *GCCA-0103: MOVEMENT OF DRY STORAGE CASKS OVER SPENT FUEL, FUEL IN THE REACTOR CORE, OR SAFETY-RELATED EQUIPMENT*
- *GCCA-0104: INACCURACY OF DIAGNOSTIC EQUIPMENT FOR MOTOR-OPERATED BUTTERFLY VALVES*
- *GCCA-0105: CROSS-TIED SAFETY INJECTION ACCUMULATORS*
- *GCCA-0106: HYDROGEN GAS IGNITION DURING WELDING OF A VSC-24 MULTI-ASSEMBLY SEALED BASKET*

GCCA-0001: ASSESSMENT OF CONDITION OF SAFETY-RELATED STRUCTURES AND CIVIL ENGINEERING FEATURES

TAC No.: M87093 **Contact:** R.A. Benedict

Description: Nuclear power plant structures are designed to withstand low-probability natural phenomena and reactor accident loadings and are constructed utilizing stringent quality control requirements. They are robust and have not been subjected to the low-probability challenges for which they were designed

except on two occasions: the accident at TMI-2 and the fierce wind loadings imposed by Hurricane Andrew on the structures of Turkey Point 3 & 4. Structures subjected to the loadings from these events withstood the loads without appreciable damage. However, information on the failures of non-nuclear structures, such as highway bridge-

decks and parking garages, indicates that the age-related degradation of well-designed and properly constructed concrete structures could weaken them sufficiently to cause them to fail without being subjected to abnormal loadings. Several incidents of age-related degradation of nuclear structures have been reported.

The objectives of this study were to: (1) review the known information on the degradation of structures and assess their conditions with respect to their safety functions; (2) make observations as to whether these safety functions are maintained for the life of the plant; and (3) provide information that could be useful for the improved design and construction of structures of future reactors.

Originating Document: NUREG/CR-4652, "Concrete Component Aging and Its Significance Relative to Life Extension of Nuclear Power Plants," September 1986.

Regulatory Assessment: Six vintage plants were visited to collect information for assessing the existing condition and past performance of structures and civil engineering features such as Seismic Category I buildings, tanks, cable trays, conduit and equipment supports, underground structures, water intake structures and anchorages, and fuel racks. Structural components included reinforced concrete, structural steel, and masonry walls.

The review concluded that most of the civil/structural plant features have performed very well. Some structures/ components showed signs of aging degradation. The 11 degradation categories were listed and rated for each plant. The ratings were not judgments about the current overall safety of these specific plants.

For license renewal applications, the ratings provide guidance for identifying the types of degradation that may require detailed review during the license renewal process and note the desirability of regular inspections and maintenance of particular structures and equipment. For future plants, the ratings provide guidance for

identifying the types of potential degradation that need to be addressed during the licensing process. The report presents the results of the staff's study of the six plants; it does not indicate a need for immediate action with respect to operating plant safety.

Resolution: Issuance of NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures."

Completion Date: 08/28/95

GCCA-0002: ENVIRONMENTAL LICENSING AND REGULATORY CONCENTRATIONS IN BUILDING WAKES

TAC No.: M87875 **Contact:** C.V. Hodge

Description: GDC 19 of 10 CFR 50, Appendix A, sets forth the requirements for control rooms at nuclear power plants. This criterion states that "[a]dequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures

is excess of 5 rem." Computational means for assessment of licensee compliance with this requirement are lacking.

Originating Document: GI 83, "Control Room Habitability."

Regulatory Assessment: Beginning in the mid-1980s, RES sponsored development of a computer code on diffusion in the vicinity of buildings. PNL completed development of a computer code ARCON95, described in NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes," May 1995. This code uses hourly meteorological data and recently developed methods for estimating dispersion in the vicinity of buildings to calculate relative concentrations at control room air intakes that would be exceeded no more than 5% of the time. These concentrations are calculated for averaging periods ranging from one hour to 30 days in duration. Relative concentrations calculated by ARCON95 are significantly lower than concentrations calculated using the currently accepted procedure when winds are less than two meters/second. For higher wind speeds, ARCON95 calculates about the same concentrations as the current procedure.

Resolution: EMCB is tracking RES activity on GI-83 on control room habitability but has no direct activities associated with the GI; therefore, the TAC No. was closed.

Completion Date: 07/21/95

GCCA-0003: RRG, 50.54(P) GUIDANCE

TAC No.: M88788 **Contact:** J.W. Shapaker

Description: On 01/04/93, the EDO established a Regulatory Review Group (RRG) to conduct a review of power reactor regulations and related processes, programs, and practices. One resulting RRG recommendation was to change the current practice that enables licensees to make changes to their security plans without prior NRC approval, i.e., using the provisions of 10 CFR 50.54(p).

Originating Document: RRG Final Report, Volume 2 - Regulations, Section 2.3, Position Paper 2.3.18, "Security," dated August 1993.

Regulatory Assessment: The plan developed by the staff for implementing the RRG recommendation was not to change the regulations, but to clarify the process by providing revised screening criteria that would ensure consistency of security plan changes without prior NRC approval.

Use of the revised screening criteria would allow licensees to reduce certain commitments that have exceeded regulatory requirements or published guidance if the overall effectiveness of the plan is not reduced. Each issue is reviewed against the overall assurance levels contained in the plan and not against specific individual changes. Latitude has always existed in that improvements in one area of the program may offset reductions in other areas. Overall assurance levels of the plans must be maintained and this clarification is not intended to reduce plan commitments to levels less than the overall high-assurance objectives stated in 10 CFR 73.55(a).

NRC has expected that licensees would judiciously make the proper determination regarding 10 CFR 50.54(p) changes and implement those changes as permitted by the regulations. This position was the original intent of the Commission and remains so today. The NRC believes that, with the use of the revised screening criteria and expertise of the licensee staff, licensees should implement changes made pursuant to 10 CFR 50.54(p) without prior NRC approval.

Resolution: Issuance of GL 95-08, "10 CFR 50.54(p) Process for Changes to Security Plans Without Prior NRC Approval," dated 10/31/95.

Completion Date: 10/31/95

GCCA-0004: RELOCATION OF SELECTED TS REQUIREMENTS RELATED TO INSTRUMENTATION (GL)

TAC No.: M90014 **Contact:** J.W. Shapaker

Description: Licensees that have not converted, or are not in the process of converting, to the improved STS may request a license amendment to relocate selected instrumentation requirements from their TS. This line-item TS improvement was developed in response to TS amendments proposed by licensees and NRC TS improvement initiatives.

Originating Documents: NRC Region III Morning Report 3-95-0019, dated 02/22/95; 10 CFR 50.72, Event Notification 28926, dated 06/11/95.

Regulatory Assessment: Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TS as part of the license. In 10 CFR 50.36, the Commission established the regulatory requirements related to the content of TS; however, the regulation does not specify the particular requirements to be included in TS. The NRC developed criteria, as described in the "Final Policy

Statement" (Federal Register Notice 58 FR 39132), to determine which of the design conditions and associated surveillances should be located in the TS as limiting conditions for operation. Four criteria were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36 (Federal Register Notice 60 FR 36953):

- (1) installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the RCPB;
- (2) a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

(4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

Implementation of these criteria may cause some requirements to be moved out of existing TS to documents and programs controlled by licensees. In this regard, GL 95-10 addresses the relocation of selected TS requirements related to instrumentation as a result of applying the 10 CFR 50.36 criteria; explicit guidance is given to licensees for submitting a proposed license amendment to accomplish this.

Resolution: GL 95-10, "Relocation of Selected Technical Specifications Requirements Related to Instrumentation," dated 12/15/95.

Completion Date: 12/17/95

GCCA-0005: BWR - SCRAM SOLENOID PILOT VALVE PROBLEMS

TAC No.: M90285 **Contact:** D.L. Skeen

Description: Slow scram times were noted at some BWR plants as a result of the fluoroelastomer material used in ASCO HV-176-816 (T-ASCO) scram solenoid pilot valves (SSPVs). An apparent change in the material supplied to ASCO by a sub-supplier caused the material to soften and allowed the SSPV to stick immediately after being deenergized. The sticking SSPV caused a small delay in the scram time of the affected control rod.

Originating Document: Grand Gulf Event Notification 26996, dated 03/27/94.

Regulatory Assessment: A relatively few number of plants were affected. All four of the BWR-6 plants, one BWR-5 plant and 2 BWR-4 plants use the T-ASCO valves in their scram systems. However, only two of these plants have reported slow scram times that were attributed to the fluoroelastomer. Discussions with GE indicated that only the BWR-6 plants are more likely to be affected because of the faster scram requirements for that vintage plant. The delays seen were in the 10 to 400 millisecond range. Even though only two plants reported slow times, the other plants were made aware via the nuclear network entries made by those two plants and GE SIL No. 591, "Delayed SCRAM Solenoid Pilot Valve Operation," issued on 05/12/95.

Resolution: Based on the small number of potentially affected plants, the improved specifications for the elastomers developed by GE and ASCO as of April 1995, and the fact that GE identified the potential problem to licensees in a SIL, the Events Assessment Panel canceled the development of an information notice on 07/18/95.

Completion Date: 07/18/95

GCCA-0006: SHIFT STAFFING ISSUE FOLLOWUP

TAC No.: M91163 **Contact:** N.K. Hunemuller

Description: On 11/26/91, the NRC issued IN 91-77, "Shift Staffing at Nuclear Power Plants," to alert licensees to the problems that could result from inadequate controls to ensure that shift staffing is sufficient to accomplish all functions required by an event. However, after IN 91-77 was issued, event follow-up inspections indicated that problems involving shift staffing and task allocation continued to occur. As a result, the NRC continued with further research in this area. This research included an RES project to address the adequacy of minimum shift staffing levels through a shift staffing study encompassing all licensee staff initially needed during an event.

Originating Document: NRR Task Action Plan, "Nuclear Power Plant Shift Staffing," dated 04/13/95.

Regulatory Assessment: The licensees surveyed generally staffed to levels greater than those required by either the regulations or their plant-specific TS for both licensed and non-licensed personnel. Nevertheless, the results of the research project provide several insights into areas which could impact the ability to accomplish safety functions following an event.

Resolution: IN 95-48, "Results of Shift Staffing Study," dated 10/10/95, was issued to inform licensees of the results of the NRC's study conducted as part of the RES project to address the adequacy of minimum shift staffing levels at nuclear power plants.

Completion Date: 10/10/95

GCCA-0007: LESSONS LEARNED FROM OPERATIONAL SAFEGUARDS RESPONSE EVALUATIONS

TAC No.: M91231 **Contact:** E.J. Benner

Description: At 6:53 a.m. on 02/07/93, an intruder drove into the TMI site entrance, continued past the guard house, and crashed through Gate I of the protected area. The vehicle proceeded to crash through a turbine building roll-up door and came to a stop 63 feet inside the turbine building, enveloped by the roll-up door. Control room personnel responded by implementing emergency response procedures, including locking control room fire doors, and classifying the event as a Site Area Emergency (SAE) at 7:05 a.m. Security staff responded by posting security personnel to intervene at pre-designated vital areas, confirming vital area integrity and, with

the aid of offsite responders, conducting an assessment of and search for the intruder. TMI security personnel found and apprehended the unarmed intruder at 10:57 a.m. The intruder was located at the bottom of the turbine building in a small space under condenser piping and offered no resistance. The intruder was questioned onsite by Pennsylvania State Police then escorted offsite in custody. The U.S. Army explosives ordinance

disposal unit completed a detailed search confirming that no explosives were present. Upon visually inspecting plant equipment, verifying plant parameters, and confirming that safety systems were available, the licensee terminated the SAE at 4:25 p.m.

Originating Document: Event Notification 25035, dated 02/07/93.

Regulatory Assessment: Due to the vehicle intrusion event at TMI, the EDO assigned NRR, NMSS, AEOD, RES, and Region I responsibility for taking generic actions resulting from the investigation of the unauthorized forced entry of the protected area at TMI on 02/07/93. An action plan was developed under TAC No. M40031 to provide generic guidance on unauthorized forced entry.

The action plan determined that lessons learned from the Operational Safeguards Response Evaluations should be issued to licensees in the form of an IN; TAC No. M91231 was for development of the IN. Because of the sensitive nature of the information, it was subsequently determined that an IN was an inappropriate method of informing licensees. Instead, the decision was made to send individual letters to all plant security managers with the safeguards-classified information attached.

Resolution: Issuance of individual letters to all plant security managers with safeguards-classified information attached. (See Accession No. 9509260227 for a sample letter.)

Completion Date: 12/12/95

GCCA-0008: AIR ENTRAINMENT IN TERRY TURBINE LUBRICATING CONTROL OIL SYSTEM

TAC No.: M91307 **Contact:** D.L. Skeen

Description: Entrainment of air into the oil system that serves the dual purpose of lubrication of the turbine bearings and hydraulic speed control was identified as a problem for AFW systems in PWRs and RCIC systems in BWRs. The HPCI systems in BWRs were not susceptible to this problem because they are equipped with separate oil pumps that preclude the air entrainment. The air entrainment presents a potential problem of overheating of a turbine bearing or turbine speed fluctuations.

Originating Document: Pilgrim Event Notification 27621, dated 08/03/94.

Regulatory Assessment: Air entrainment could present a potential problem to both AFW and RCIC systems. The vendor stated that a 4-hour test run at the factory and initial start-up testing at the plant usually identify the problem and a larger drain line and/or a vent at the bearing housing can be installed to prevent any further problems with air entrainment. Despite the vendor claims, reports from at least three plants experiencing air entrainment problems prompted the NRC to issue an IN.

Resolution: IN 94-84, "Air Entrainment in Terry Turbine Lubricating Oil System," was issued on 12/02/94 to address the air entrainment issue, while the need for a GL was being considered. Subsequent events involving binding of the governor valve stem and blockage of the turbine drain pots prompted the NRC to consider Terry Turbine reliability as a whole and TAC No. M92636 was opened to track the long-term follow-up of Terry Turbine activities. Since the air entrainment concern was included in the long-term follow-up, TAC No. M91307 was closed based on a memorandum from G. Holohan to B. Grimes dated 07/21/95.

Completion Date: 07/21/95

GCCA-0009: WRONG REPLACEMENT PARTS RELIEF VALVES AND REFUELING MAST

TAC No.: M91399 **Contact:** J.L. Birmingham

Description: This proposed IN discussed three instances in which the installation of incorrect replacement parts resulted (or may have resulted) in equipment damage or improper performance. The instances occurred at River Bend on 09/21/94, at Cooper on 08/30/94, and at ANO-2 in July 1990.

Originating Document: Memorandum from W.P. Ang to A.E. Chaffee, signed 05/15/95, proposing that an IN be issued to address the installation of incorrect replacement parts.

Regulatory Assessment: The probable adverse consequences of using incorrect replacement parts in safety or non-safety-related equipment is a concern that is generally well understood within the nuclear industry.

Although the concern for the use of incorrect parts in equipment is valid, the information in the proposed notice did not provide a new or different understanding of the issue. The concern that licensees use correct parts

during equipment repair or replacement is best addressed by licensee audit programs and by NRC inspection of licensee programs for maintenance. Therefore, an IN on this issue is not necessary at this time.

Resolution: The need for the proposed IN was discussed with representatives of Region IV (the originating group) and agreement was reached that the proposed notice was not needed at this time. The proposed notice was cancelled 07/11/95 via an E-mail message from W.P. Ang, Region IV.

Completion Date: 07/20/95

GCCA-0010: SPENT FUEL POOL OVERFLOW INTO VENTILATION SYSTEM

TAC No.: M91400 **Contact:** N.K. Hunemuller

Description: A draft IN from Region II discussed the results of follow-up reviews of an event at Brunswick-2 when water from the spent fuel pool overflowed into the ventilation system and drained onto several elevations of the reactor building floor.

Originating Document: Memorandum from B.A. Boger to B.K. Grimes, "Proposed Information Notice - Spent Fuel Pool Overflow," dated 11/21/94.

Regulatory Assessment: This single plant-specific event did not raise to the level of concern requiring the issuance of an IN. NRR Task Action Plan, "Generic Spent Fuel Storage Pool," addresses spent fuel pool safety concerns.

Resolution: After management review, the IN was cancelled based on the 07/21/95 memorandum from G.M. Holahan to B.K. Grimes, "Review of Generic Communications and Compliance Activities." The proposed IN failed to meet the threshold for regulatory action based on its safety significance and the number of affected plants. However, spent fuel pool safety concerns are being addressed in the NRR Task Action Plan, "Generic Spent Fuel Storage Pool."

Completion Date: 07/21/95

GCCA-0011: IPEEE FOR SEVERE ACCIDENT VULNERABILITIES

TAC No.: M91401 **Contact:** J.W. Shapaker

Description: In 1991, GL 88-20, Supplement 4 was issued requesting all licensees to perform an IPEEE to find plant-specific vulnerabilities to severe accidents caused by external events, including seismic events, and report the results to the NRC. A companion document, NUREG-1407, provided procedural and submittal guidance

for the IPEEE. Review level earthquakes (RLEs) and the review scope were defined by the staff for all U.S. sites; plants in the central and eastern U.S. were assigned to appropriate review categories (plant bins) primarily according to a comparison of available seismic hazard results.

The hazard results used in the binning process included those published in 1989 by the LLNL (NUREG/CR-5250) and the EPRI (NP-6395-D). NRC established the bins because of the large inherent uncertainties in the probabilistic estimation of seismic hazard. Using this approach, the staff compared the relative seismic

hazard of the 69 central and eastern U.S. plant sites, and assigned each plant to one of four bins for the seismic margins method (Reduced-Scope, 0.3g Focused-Scope, 0.3g Full-Scope, and 0.5g bin).

In 1994, based on a re-elicitation of LLNL ground-motion and seismicity experts, the staff published revised seismic hazard results in NUREG-1488. The new LLNL mean hazard estimates were lower than the 1989 LLNL results but higher than the EPRI estimates. In a letter from W. Rasin (NEI) to A. Thadani (NRC) on 04/05/94, NEI advocated that most focused-scope plants should instead perform reduced-

scope studies as part of the seismic IPEEE, based on the revised hazard estimates. NEI also stated that each licensee is responsible for proposing the most cost-effective program to satisfy the seismic IPEEE request consistent with the level of seismic hazard at the specific site. As a result, seven licensees informed the NRC of their intent to revise their IPEEE commitments.

These developments prompted the NRC to systematically revisit the seismic IPEEE program rather than deal with each licensee individually. In IN 94-32, the staff stated that it would review LLNL's revised seismic hazard estimates and determine the appropriateness of revising the seismic IPEEE scope.

NRC contracted with Energy Research, Inc. (ERI) to do a seismic revisit study to determine whether consideration of the new LLNL seismic hazard estimates would: (1) significantly change the original binning

results; and (2) warrant adjusting the seismic scope and guidelines of the seismic IPEEE review. The latter effort would also require the determination of how the scope should be modified and the justification of such modifications. ERI completed the study and submitted two reports (ERI/NRC 94-502 and ERI/NRC 94-504). The

staff subsequently held a public workshop to discuss these reports, present comments from a peer review group, determine issues to be addressed, and solicit public input for developing the staff position on the seismic scope modification.

Originating Document: IN 94-32, "Revised Seismic Hazard Estimates," dated 04/29/94.

Regulatory Assessment: The NRC staff evaluated the ERI reassessment reports, the peer review group comments, the NEI white paper and comments received at and after the workshop. The staff concluded that:

(1) licensees may use the revised LLNL seismic hazard estimates instead of the 1989 LLNL seismic hazard estimates in the seismic PRA; and (2) the scope of the seismic IPEEE may be modified for all focused-scope and full-scope plants by eliminating the need to calculate the capacity of certain generally rugged components, or certain site effects that would not be significant sources of contributors to seismic severe accident risk or would not result in cost-beneficial improvements. The justification for this reduction in the seismic review scope is that the perceived seismic hazard estimates and associated risks have decreased. However, the examination process for the modified seismic IPEEE remains the same as the process described in Supplement 4 to GL

88-20 and NUREG-1407.

Resolution: Issuance of GL 88-20, Supplement 5, "Individual Plant Examination of External Events for Severe Accident Vulnerabilities," dated 09/08/95.

Completion Date: 09/08/95

GCCA-0012: SURRY VENTILATION FILTER ISSUE

TAC No.: M91438 **Contact:** R.A. Benedict

Description: Cleaning the secondary side of a steam generator was performed at Surry Unit 2 using chemicals which, when vented through the charcoal adsorbers of the ventilation system, reduced significantly the iodine- removal efficiency of the charcoal. The licensee was not aware that some of the chemicals used could degrade the charcoal.

Originating Document: Oral report from G. Hubbard, followed by NRC Inspection Report 50-280/94-21.

Regulatory Assessment: Exposure of charcoal to chemical compounds can result in a degradation of the charcoal. In the event of a nuclear accident, the degraded charcoal in the ESF filtration systems may perform at an efficiency significantly less than that assumed in the DBA dose assessments in the staff's safety evaluation. Chemical cleaning is performed on a frequent basis at most PWRs, either every or every other refueling outage. The IN is needed to inform the industry of the potential problem associated with chemical cleaning of components.

Resolution: Issuance of IN 95-41, "Degradation of Ventilation System Charcoal Resulting from Chemical Cleaning of Steam Generators," dated 09/22/95.

Completion Date: 09/30/95

GCCA-0013: COMMON MODE FAILURE OF COPES VOLCAN PORVs

TAC No.: M91446 **Contact:** E.J. Benner

Description: Haddam Neck has two Copes-Vulcan PORVs both of which failed a surveillance test on

02/19/94 due to a leak in the diaphragm assembly. During the 1993 refueling outage which ended in July 1993, the licensee replaced the PORV diaphragms with a new type made of different material and of a changed shape. A lubricant was needed to help install the diaphragms due to the changed shape. The lubricant was believed to have allowed some extrusion of the diaphragm from between the base and the cover away from the bolt holes.

The extrusion caused tears at several bolt holes and allowed the bolts to loosen over time and air to escape. An evaluation of previously reported diaphragm failures disclosed other failure mechanisms.

Originating Document: Event Notification 28923.

Regulatory Assessment: PORVs are designed to remain operable for 30 hours during a DBA and to be capable of four valve strokes during feed-and-bleed scenarios. PORVs may be supplied air to open from either

non-safety-related air compressors or an emergency air accumulator. The non-safety-related air compressors cannot be assumed to operate during an accident since the air compressors are not designed to operate in the harsh environment that may exist during the time when the PORVs may be required to open. Once the compressed air in the emergency air accumulator has been depleted, the PORVs fail closed, and core cooling capability is lost if auxiliary feedwater and main feedwater are not available for steam generator cooling. This

concern is limited to those plants with high head injection pumps incapable of lifting pressurizer safety valves, for which extended inoperability of the PORVs may be highly risk significant.

The issue of pressurizer PORV reliability has been addressed in GL 90-06, "Resolution of Generic Issue 70,

'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,'" in which the staff requested licensees to include the PORVs and PORV control air system in their ASME Section XI IST Program. An IN was issued to alert licensees to PORV failure mechanisms which may render PORVs degraded in significantly shorter time than common IST surveillance frequencies, thereby resulting in unexpected inoperability of the PORVs following certain DBAs.

Resolution: IN 95-34, "Air Actuator and Supply Air Regulator Problems in Copes-Vulcan Pressurizer Power- Operated Relief Valves," dated 08/25/95.

Completion Date: 08/25/95

GCCA-0014: DEFICIENCIES IDENTIFIED DURING ELECTRICAL DISTRIBUTION SYSTEM INSPECTIONS

TAC No.: M91448 **Contact:** S.S. Koenick

Description: This IN supplement represents the closeout of the electrical distribution system functional inspection. This supplement provides additional information on deficiencies identified during the functional inspections. Furthermore, this supplement provides references for previously identified deficiencies.

Originating Document: TI 2515/107, "Electrical Distribution System Functional Inspection," issued 10/19/90.

Regulatory Assessment: TI 2515/107 was issued in response to electrical distribution system deficiencies identified by various multi-discipline inspection efforts. The TI provided instruction that allowed for a consolidated and consistent inspection of the utilities' electrical distribution system. Any safety concerns identified during the particular inspections would be appropriately dispositioned in the respective inspection reports. The IN and the prior supplements allow for a forum to share this information with the rest of the industry.

Resolution: IN 91-29, Supplement 3, "Deficiencies Identified During Electrical Distribution System Functional Inspections," was completed in the spring of 1994 and issued on 11/22/95.

Completion Date: 11/22/95

GCCA-0015: SEISMIC ADEQUACY OF THERMO-LAG PANELS

TAC No.: M91531 **Contact:** T.J. Carter

Description: A concern regarding seismic adequacy of fire barriers that use Thermo-Lag was raised by the Nuclear Information and Resource Service in a 07/21/92 petition [See Accession No. 9208280125]. The staff, based upon available information on physical properties of Thermo-Lag and analysis by a consultant of Thermal Science, Inc., concluded that the concern was not credible [See Accession No.

9302110146]. A subsequent submittal containing results of simulated seismic test and mechanical properties tests related to the use of Thermo-Lag fire barrier material at Watts Bar utilized significantly lower mechanical properties values compared to those used by the consultant. As a result, the NRC became concerned that the actual properties might be sufficiently different and that performance of Thermo-Lag might not be acceptable.

Originating Document: A letter from TVA to the NRC on 11/11/94 [See Accession No. 9411250234].

Regulatory Assessment: In December 1994, the staff sent followup letters to GL 92-08, "Thermo-Lag 330-1 Fire Barriers," to request additional information, including specifics on the mechanical properties. Licensees have been alerted to the concern and action has been taken to ultimately resolve the concern. This information notice clearly explains the concern and indicates how the concern will be resolved.

Resolution: IN 95-49, "Seismic Adequacy of Thermo-Lag Panels," was issued on 10/27/95 to inform licensees of concerns involving actual properties of Thermo-Lag panels.

Completion Date: 10/27/95

GCCA-0016: CAPABILITY OF OFFSITE POWER DURING DESIGN BASIS EVENTS

TAC No.: M91533 **Contact:** T. Koshy

Description: On 01/05/95, Palo Verde reported to the NRC that, under certain offsite power system grid voltage levels, the performance of the automatic accident mitigation would be uncertain. This problem was introduced through a licensee resolution to a voltage regulation problem. The automatic loading of the ECCS could be subjected to a "double sequencing" as a result of the anticipated system performance. This problem appeared to be a plant-specific concern that was discovered by the licensee in the design bases reconstitution effort. The later discussions revealed the possibility of gradual changes in switchyard voltage profile and load changes within the plant as potential causes for this scenario.

When grid voltage is at a minimum acceptable level, the addition of safety injection loads to the bus leads to further drop in grid voltage and results in a loss of AC power. No design requirements were added into the regulation since it was not considered to be a credible event in the early evolutions of power system design. The compensatory actions that can be taken are simple in nature when the control room operators are sensitized to this potentially remote vulnerability. On 07/08/95, Diablo Canyon 1 & 2 reported a similar problem.

Originating Document: 10 CFR 50.72, Event Notification 28210 and 29168.

Regulatory Assessment: When the NRC was made aware of this scenario, the staff evaluated the Palo Verde corrective actions. The short-term corrective action was to prevent automatic loading of safety injection loads to a marginally acceptable offsite power source. The long-term solution is to appropriately evaluate the capability of the electrical system to support emergency loads and revise the setpoints for electrical bus transfer and emergency diesel generator loading.

Even though this scenario can cause unavailability of certain safety systems, the probability of such occurrence is very low as it would require marginally acceptable grid voltage and a valid demand for safety injection.

Therefore, an information notice was chosen as the suitable regulatory response to sensitize the industry about this potential vulnerability, and to reinforce the need for the periodic review of electrical system capability to support accident mitigation.

Resolution: IN 95-37, "Inadequate Offsite Power System Voltages during Design Basis Events."

Completion Date: 09/07/95

GCCA-0017: POTENTIAL FOR LOSS OF AUTOMATIC ESF ACTUATION

TAC No.: M91534 **Contact:** E.N. Fields

Description: The Salem and Diablo Canyon licensees reported a condition that could have resulted in the failure of one or both trains of the SSPS during a seismic event or a main steamline break in the turbine building (TAC No. M91479). IN 95-10 was issued to inform other licensees of the findings by the Diablo Canyon and Salem licensees. Both licensees attempted to implement a design change that they felt would eliminate the design vulnerability. The Diablo Canyon licensee was successful in making the design change; however, the Salem licensee encountered numerous problems in attempting the design change.

Supplement 1 to IN 95-10 provided details on the problems the Salem licensee experienced, including inadvertently de-energizing the source range high voltage block signal and the problems encountered in attempting to remove the SSPS logic matrix power supplies.

Supplement 2 to IN 95-10 was issued on 08/11/95 primarily to communicate the Salem licensee's

corrective actions with respect to the surveillance and maintenance practices for the SSPS. The licensee implemented these corrective actions in an effort to improve the reliability of the SSPS. It appeared worthwhile to apprise the industry of these actions.

Originating Document: 10 CFR 50.72, Event Notification 28318.

Regulatory Assessment: Shortly after the NRC was made aware of the SSPS design deficiency and the problems encountered by the Salem licensee in attempting corrective actions, affected licensees were apprised of their potential susceptibility to the problem. Licensees were notified through direct contact by NRC project managers, through **W** notification, and through IN 95-10 (and Supplement 1). Therefore, there is no immediate

safety concern with regard to the timing of the issuance of this IN Supplement 2 and the overall timing of the staff's efforts in addressing the concerns that have surfaced from this event has not been inappropriate.

Resolution: IN 95-10, Supplement 2, "Potential for Loss of Automatic Engineered Safety Features Actuation," dated 08/11/95, and Memorandum from B. Boger to W. Russell dated 04/11/95.

Completion Date: 08/16/95

GCCA-0018: POTENTIAL FOR MOV FAILURE - STEM PROTECTION PIPE CHANGES

TAC No.: M91624 **Contact:** T.A. Greene

Description: A RHR MOV at Cooper failed to closed on demand while the RHR trains were being switched. Licensee reviews showed that the stem protector pipe on the valve actuator had threaded into the MOV housing and interfered with the stem nut rotation. The motor was damaged during an attempted opening stroke. The stem protector pipe had previously been replaced by the licensee. The replacement stem protector pipe was manufacture by the licensee. However, it was not constructed to the same tolerance of the original. Specifically, the length of threaded portion of the protector pipe was too long when manufactured. The extended threads allowed the pipe to be threaded to the point where it interfered with the stem nut locknut. Also, actions (such as staking) were not taken to prevent threading of the stem protection pipe into the valve actuator housing.

Originating Document: Morning Report 4-95-0014.

Regulatory Assessment: A stem protector pipe may be attached to an MOV through the housing cover to prevent debris from entering the stem/actuator interface area. To keep the stem protector pipe from interfering with actuator operation (specifically rotation of the stem nut and its locknut), the threads on the pipe may be restricted to a certain length. Another option is to stake the threads on the stem protector pipe at a specific location. If neither of these precautions is taken, the stem protector pipe may thread sufficiently into the actuator housing to interfere with the rotation of the stem nut locknut. Additional torque may be required to operate the valve, which may cause the torque switch to trip prematurely, motor thermal overload devices to activate, or the motor to be damage on high torque demand.

Resolution: At Cooper, the licensee long-term solution was to stake the extended threads in all of the stem protector pipes of safety-related MOVs. IN 95-31, "Motor-Operated Valve Failure Caused by Stem Protector Pipe Interference," was issued on 08/09/95 to address the generic concern.

Completion Date: 08/09/95

GCCA-0019: UNANTICIPATED AND UNAUTHORIZED MOVEMENT OF FUEL

TAC No.: M91642 **Contact:** C.V. Hodge

Description: IN 94-13 implied a broader applicability of training requirements of 10 CFR 50.120 to contract personnel than is the case. In October 1994, DRCH/NRR proposed a supplement to IN 94-13 to clarify confusion of the original notice concerning application of 10 CFR 50.120. The descriptive language in the original notice implied a broader applicability to contract personnel than is the case. In addition, in March 1995, there were contractor mistakes at Hatch that resulted in a dropped core shroud bolt that punctured the spent fuel pool

liner, dropped stellite bearings on the transfer canal floor that produced an uncontrolled high radiation area, and placement of excessively contaminated items outside of radiologically controlled areas.

Originating Document: IN 94-13, "Unanticipated and Unintended Movement of Fuel Assemblies and Other Components Due to Improper Operation of Refueling Equipment."

Regulatory Assessment: On 03/31/95, DRCH concurred in authorizing publication of a supplement to IN 94-13 to address the original contract personnel applicability question and the Hatch events as additional examples

of problems associated with the control of contract personnel activities and inadequate oversight of refueling operations.

Resolution: The supplement was issued on 11/28/95.

Completion Date: 11/28/95

GCCA-0020: FRAUDULENT COMMERCIAL GRADE CERTIFICATE OF COMPLIANCE

TAC No.: M91643 **Contact:** C.V. Hodge

Description: In September 1989, Southern Testing Services contracted with Relay Specialties, an authorized Agastat relay distributor, to provide: (1) 32 Agastat commercial-grade Model 7022AC relays; (2) Agastat

relay sockets, P/N 700137; and (3) a "certificate of conformance for [the relays] from Agastat." In April 1991, during an NRC inspection at the Southern Testing Services facility, an NRC inspector became suspicious of an Amerace certificate of compliance because it did not appear to have the correct Amerace facility address.

During subsequent discussions between NRC and Amerace personnel, the NRC inspector found that the certificate of compliance had not been issued by Amerace, nor was the signature on the certificate of compliance that of an Amerace employee. Amerace stated that the correct Amerace facility address on a certificate of compliance for that particular time period was 530 W. Mt. Pleasant Avenue, Livingston, New Jersey. In contrast, the address shown on the attached certificate of compliance is 190 Lincoln Highway, Edison, New Jersey. The NRC inspector examined one of the Agastat relays that Southern Testing Services had used as a test specimen during the commercial-grade component testing and noted that the label affixed appeared to be a label used by the Control Component Supply company for relays that had been field-modified. Control Component Supply was an authorized distributor of Amerace relays and related components.

Originating Document: Special Inspection Report 99900289/91-01.

Regulatory Assessment: Agastat relays are used in numerous safety-related applications in nuclear power plants. Because Amerace supplied these relays as commercial-grade components, the whole nuclear power industry could theoretically procure them and dedicate them to safety-related service.

Resolution: RVIB drafted an IN. Due to the nature of the concern, concurrence in the IN was needed from OGC who asked questions regarding the development of additional events since the inspection. The IN was cancelled based on the age of the issue and the lack of any additional occurrences since the inspection.

Completion Date: 07/20/95

GCCA-0021: CHATTER OF ITT BARTON 288A AND 289A DIFFERENTIAL PRESSURE TRANSMITTERS

TAC No.: M91645 **Contact:** C.V. Hodge

Description: ITT Barton is a supplier of basic components used in safety-related applications in nuclear power plants. The company has supplied Models 288A and 289A differential pressure (**dp**) indicating switches with certification to IEEE-323/344 qualifications. These include no switch contact chatter during seismic loadings of up to 12g for monitored **dp** different from setpoint by at least 10% full scale. In 1994, unrelated testing revealed potential concerns regarding contact chatter. As a result, additional testing of qualified configurations of these products was performed at Wyle Laboratories in Norco, CA, during the first week of 1995.

Originating Document: Part 21 Report from ITT Barton on 10/05/94 [Accession No. 9411010149].

Regulatory Assessment: The safety significance and generic implications depend on how the transmitter is used. The contact chatter at 12g has reasonable probability of occurrence if the instrument is located at a high elevation and the plant is built on soil rather than rock. An SSE is usually characterized by a smaller acceleration (0.2g on the east coast, 0.5g on the west coast), but the mechanical parameters displacement, velocity, and acceleration may be amplified for various frequencies of oscillation. (Consider the mechanical model of a mass coupled through a damped spring to an oscillating base.) As explained in the Part 21 report dated 01/19/95, the chatter does not occur when the monitored **dp** is different from the setpoint by 10% full scale. The report also notes that "testing has again verified that these products

are capable of surviving earthquake loadings up to 12g with no degradation in structural or pressure boundary integrity and no degradation in functionality with regard to pointer accuracy, switch setpoint change and switch deadband." The NRC is concerned with this report because of QA aspects of safe operation of nuclear power plants. This vendor has evidently provided production models of this instrument, samples of which did not perform well on qualification tests. It follows that a production control problem may exist. No event reports have been received similar to this Part 21 report. This report relates to GI

A-46, "Seismic Qualification of Equipment in Operating Plants," and GI 114, "Seismic-Induced Relay Chatter," identified in March 1985. In the event of an earthquake of sufficient magnitude (g level) that causes relay chatter, combined with simultaneous loss of offsite power and potential misalignment of equipment, instrumentation, and circuit breakers, core damage may result, absent corrective operator action.

Resolution: In a subsequent Part 21 report dated 02/15/95, the vendor supplied a list of customers. RVIB reviewed the engineering report and conducted a vendor inspection. The following excerpt from Inspection

Report 99900113/95-01, pp. 5-6, Sec 3.6, demonstrates that this concern is closed because corrective vendor action was sufficient.

"Barton began an engineering evaluation of mild environment equipment qualification in late 1994 as a result of a licensee group audit. During review of 1980 and 1986 seismic test reports, Barton determined that switch chatter may have occurred that was not detected. The specific instrumentation used to monitor contact chatter was not identified in the test reports, but was suspected of being an incandescent lamp. Barton conducted additional seismic tests early in January 1995, using instrumentation capable of measuring contact chatter as rapid as

two milliseconds. The new testing showed that higher g levels, and setpoints very close to actual parameter values, produced the most chatter; there was no chatter at 4g. Barton provided all affected customers with a table showing the duration of contact chatter as a function of g level and the proximity of the trip setpoint to the actual differential pressure value. The NRC inspector discussed this with Barton. The review of data from earlier seismic tests, and conducting additional tests, revealed a possible concern that had gone unnoticed for several years. The inspector considered Barton's activities including notifications to be acceptable, and no further action is required."

No NRC generic response, such as an IN, is needed. Two previous INs are related to this issue: IN 85-02, "Improper Installation and Testing of Differential Pressure Transmitters," and IN 86-65, "Malfunctions of ITT Barton Model 580 Series Switches During Requalification Testing."

Completion Date: 07/20/95

GCCA-0022: FREQUENCY OF USE OF AIR-OPERATED GATE VALVES

TAC No.: M91746 **Contact:** C.V. Hodge

Description: Air-operated gate valves, utilizing Hillar actuators, in the room cooler water supply lines failed to open (safety function) during surveillance. The problem seems to be related to the frequency of operation. Hope Creek has 32 valves of this type.

Originating Documents: 10 CFR Part 50.72, Event Notification 28014, dated 11/10/94; LER 50-354/94-17, dated 12/08/94.

Regulatory Assessment: The LER explains that Hope Creek systems engineering determined a common cause (packing design). The probable root cause is packing configuration combined with a long interval between valve stroking. These valves were subject to a design change from 9 rings of Crane packing to 9 rings of Chesterton Graphfoil packing. The industry standard configuration is 4 or 5 rings of graphite packing with carbon busing. The standard packing gland torque calculations are based on 5 rings. By calculation, reduction from 9

to 4 or 5 rings reduces the dynamic packing load up to 50% and reduces the total force needed by the actuator. A contributing cause may involve use of the disk friction coefficient from the original sizing calculations. Such usage is generally non-conservative, according to findings from GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." Industry experience shows that long stationary times foster packing sticking.

Thus, static packing loads increase substantially and the total load may exceed the capability of the actuator. Also, a higher stem finish is needed for graphite packing. Safety-related stuck air valve events have been reported from Peach Bottom.

Resolution: The licensee reported that the problem related to the packing changeout was the failure to adjust the air regulator to accommodate the lower packing friction. There was some evidence that the force of the actuator drove the valve disk into the seat and then the spring force could not dislodge it within the required stroke time. Further discussions with the licensee indicated that a contributing factor, which may have been the overriding reason for failure, was that the licensee failed to install mechanical

stops on the valve. These stops prevent the valve actuator from driving the valve beyond its closed position. While valve packing issues

are generic to the industry, the main cause appears to be related to the failure to install the stops which is not a generic issue. Therefore, an IN was not deemed necessary.

Completion Date: 08/31/95

GCCA-0023: PRESSURE LOCKING AND THERMAL BINDING OF GATE VALVES

TAC No.: M91781 **Contact:** J.W. Shapaker

Description: In GL 89-10 (06/28/89), the NRC staff asked addressees to provide additional assurance of the capability of safety-related MOVs and certain other MOVs in safety-related systems to perform their safety-

related functions. In Supplement 6 to GL 89-10 (03/08/94), the NRC staff provided guidance on an acceptable approach for addressing pressure locking and thermal binding of MOVs, but did not request specific actions. During inspections of GL 89-10 programs, the NRC staff found the actions taken by licensees to address pressure locking and thermal binding of MOVs to be varied. In view of these inspection results, and the fact that most addressees were nearing completion of their GL 89-10 programs, the NRC staff determined that

issuance of a subsequent GL was warranted to request that addressees perform, or confirm that they previously performed: (1) evaluations of operational configurations of safety-related, power-operated (including motor-

air-, and hydraulically-operated) gate valves for susceptibility to pressure locking and thermal binding; and (2) further analyses and any needed corrective actions to ensure that safety-related, power-operated gate valves that are susceptible to pressure locking or thermal binding are capable of performing the safety functions within the current licensing bases of the facility.

Originating Documents: GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," dated 06/28/89; GL 89-10, Supplement 6, "Safety-Related Motor-Operated Valve Testing and Surveillance," dated 03/08/94.

Regulatory Assessment: 10 CFR 50 (Appendix A, Criteria 1 and 4) and plant licensing safety analyses require and/or commit that the addressees design and test safety-related components and systems to provide adequate assurance that these systems can perform their safety functions. Other individual criteria in Appendix A to

10 CFR Part 50 apply to specific systems. In accordance with these regulations and licensing commitments and under the additional provisions of 10 CFR 50 (Appendix B, Criterion XVI), licensees are expected to take actions to ensure that safety-related, power-operated gate valves susceptible to pressure locking or thermal binding are capable of performing their required safety functions. Supplement 6 to GL 89-10 alerted licensees to the problems with pressure locking and thermal binding in MOVs and described an acceptable approach for addressing these phenomena for MOVs, but did not request any specific actions or response from licensees.

The actions requested in GL 95-07 are considered compliance backfits, under the provisions of 10 CFR 50.109 and existing NRC procedures, to ensure that safety-related, power-operated gate valves that are susceptible to pressure locking or thermal binding are capable of performing their intended safety functions. In accordance with the provisions of 10 CFR 50.109 regarding compliance backfits, a full backfit analysis was not performed. However, the staff performed a documented evaluation which stated the objectives of and reasons for the requested actions and the basis for invoking the compliance exception. This evaluation is available in the NRC PDR [Accession No. 9508240264].

Resolution: Issuance of GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," dated 08/17/95.

Completion Date: 08/17/95

GCCA-0024: DEGRADED DECAY HEAT REMOVAL CAPABILITY VIA NATURAL CIRCULATION

TAC No.: M91805 **Contact:** J.R. Tappert

Description: The ability of steam generators (SGs) to remove decay heat via sub-cooled natural circulation may be degraded under certain plant conditions; specifically, when the RCS is vented. This limitation is not explicit in TS and some licensees have relied on SGs as one of their two redundant sources of decay heat removal when they may not have been able to fully perform this function.

Originating Document: Memorandum from E. Merschoff to B. Grimes dated 03/09/95.

Regulatory Assessment: TS generally require two methods of decay heat removal in Mode 5 with loops filled. When this is the case, they generally go on to indicate that this requirement can be satisfied by two loops of RHR or one loop of RHR and a minimum water level in the SGs. Decay heat can be removed either through the RHR system or through the SGs by natural circulation after the RCPs are secured. The heat removal mechanism with RHR is through forced circulation through the RHR heat exchanger. Heat removal with natural circulation of reactor coolant through the SGs occurs because of the differential pressure created between

the heated water in the reactor core and the cooler water in the SG tubes. This differential pressure is created through temperature differences that in turn create fluid density differences between these two locations.

During natural circulation, the SG secondary side water boils and steams off through the atmospheric relief valves or other openings that may exist during shutdown conditions. The minimum temperature at which boiling will begin in the SG is 100C [212F]. A minimum temperature differential of 28C [50F] between the RCS and the

SG secondary water is routinely used for evaluating conditions that would ensure sufficient natural circulation flow to prevent boiling in the core. The heat transfer rate across the SG tubes is less for lower RCS-to-SG secondary temperature differentials but still may be adequate to promote sufficient natural circulation and prevent core boiling. Adding the differential temperature of 28C [50F] to 100C [212F] results in a minimum RCS temperature of 128C [262F] to maintain sufficient natural circulation flow. The lowest pressure point in the RCS, at the top of the SG tubes, should therefore be maintained above the saturation pressure for 128C [262F]. If the RCS pressure at the top of the SG tubes is allowed to fall below the primary fluid saturation temperature, flashing and steam voiding may occur, interrupting or degrading the natural circulation flow path. Additionally, when system pressure is dropped with elevated water temperatures, gases may come out of solution.

When the RCS is being depressurized and cooled down, the RCPs are stopped, the RCS is depressurized and vented, and level is decreased in preparation for Mode 6 (refueling) entry. In Mode 6, both RHR trains must

be operable. During the transition from Mode 5, with no RCPs running, to Mode 6, plant conditions may exist that are not adequate to support natural circulation. The second train of RHR may need to be operable before proceeding with plant cooldown and depressurization to provide a second method for RCS cooling.

Some licensees may not clearly understand that plant conditions may degrade the SGs ability to remove decay heat by natural circulation. An IN was written to alert them to these conditions. Due to the fact that at least one train of RHR is always required, an IN was an adequate response to this issue.

Resolution: On 08/28/95, IN 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation," was issued to alert licensees to relevant plant operating experiences.

Completion Date: 08/25/95

GCCA-0025: FALSIFICATION OF ASNT CERTIFICATE BY AMERICAN POWER SERVICES

TAC No.: M91950 **Contact:** T.A. Greene

Description: American Power Service (APS) of Georgetown, Massachusetts, deliberately gave falsified American Society for Nondestructive Testing (ASNT) certificates to an NRC licensee in connection with the procurement of commercial-grade services. Northeast Nuclear Energy Company (NNECo) had contracted with APS for commercial-grade services in supplying rebabbitted bearings for the Millstone-2 reactor building closed cooling water system pump. Since APS did not have a NNECo approved QA program, the re-babbitted bearings were to be dedicated and accepted under NNECo's QA program. The dedication process included ultrasonic examination which was to be performed by NNECo as part of the receiving inspection.

During NNECo source inspection of APS, the APS president informed the NNECo Quality Assessment Services (QAS) inspector that APS had developed in-house UT capability and gave the inspector copies of what appeared to be ASNT certificates identifying three of the company's employees as UT inspectors or examiners. Upon observing that the certificates had identical serial numbers, the NNECo QAS inspector contacted ASNT and learned that ASNT had not issued the certificates in question.

Originating Document: Letter from NNECo to NRC dated 03/04/94 (Accession No. 9403150377).

Regulatory Assessment: Since APS was performing a commercial-grade service and since the acceptance of the bearings was based on UT performed by NNECo, the integrity of the bearings was not compromised and NNECo was able to justify their use in the intended application.

Resolution: The NRC Office of Investigations determined that the president of APS deliberately and improperly falsified ASNT certificates and gave them to NNECo. The NRC informed APS of its findings by

letter dated 02/14/95 (Accession No. 9502160007). The generic concern was resolved by the issuance of IN 95-45, "American Power Service Falsification of American Society for Nondestructive Testing (ASNT) Certificates."

Completion Date: 10/4/95

GCCA-0026: ADEQUACY OF EMERGENCY AND ESSENTIAL LIGHTING

TAC No.: M91952 **Contact:** N.K. Hunemuller

Description: The objective of the NRC requirements and guidelines for emergency lighting is to ensure that in the event of a fire, plant personnel can access and operate equipment and components that must be manually operated to effect safe plant shutdown. Since IN 90-69 was issued on 10/31/90, there have been several

reported events of deficient emergency lighting, involving Vermont Yankee, Cooper, Indian Point 3, Washington Nuclear 2, and Diablo Canyon. NRC inspections have also uncovered emergency lighting deficiencies.

Originating Document: Event Notification 28025, dated 11/12/94.

Regulatory Assessment: During a meeting on 04/04/95, the Events Assessment and Generic Issues Panel approved the preparation of an IN on potential problems with post-fire emergency lighting. IN 95-36, "Potential Problems with Post-Fire Emergency Lighting," was issued on 08/29/95. This notice alerts addressees to potential problems regarding emergency lighting for plant areas needed for operation of post-fire safe shutdown equipment and in access and egress routes thereto.

Resolution: IN 95-36, "Potential Problems with Post-fire Emergency Lighting," dated 08/29/95, was issued to alert licensees to relevant plant operating experiences but does not require any licensee action.

Completion Date: 08/29/95

GCCA-0027: ADDRESS CONCERNS REGARDING ASME CODE

TAC No.: M91965 **Contact:** R.A. Benedict

Description: The staff was concerned about: (1) the use of engineering judgment regarding the ASME Code;

(2) the NRC's position on the regulatory status of ASME Code interpretations prepared by ASME; and (3) inaccuracies in GL 90-05 and IN 93-21.

Originating Document: None.

Regulatory Assessment: (1) Engineering judgment cannot replace NRC requirements, whether or not these requirements contradict the ASME Code. The licensee must request relief, not use engineering judgment, when the provisions of 10 CFR 50.55a are considered. (2) NRC endorsement in 10 CFR 50.55a is limited only to those editions/addenda of the ASME Code that are specifically identified and approved. (3) The staff discussed the differences between weld overlay and weld buildup and how these may be applied to Code requirements (IN

93-21). The staff also noted that pressure boundary leaks on safety-related systems always require relief when a Code repair is not performed (GL 90-05).

Resolution: Issuance of a letter from B.W. Sheron to R.F. Reedy on 07/24/95.

Completion Date: 07/24/95

GCCA-0028: CIRCUMFERENTIAL CRACKING OF STEAM GENERATOR TUBES

TAC No.: M92004 **Contact:** E.J. Benner

Description: Since the issuance of GL 95-03, "Circumferential Cracking of Steam Generator Tubes," additional information pertaining to the tubes removed from Maine Yankee for destructive analysis has become available. In addition, the wrong title was given to NUREG-0844 in GL 95-03 as, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes." The correct title is, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." This TAC was initiated to develop an IN to correct the error in the GL and to provide additional information to the licensees.

Originating Document: GL 95-03, "Circumferential Cracking of Steam Generator Tubes," dated 04/28/95.

Regulatory Assessment: The staff issued GL 95-03 to obtain information necessary to assess compliance with requirements regarding steam generator tube integrity in light of the inspection findings

at Maine Yankee. In GL 95-03, the staff requested that utilities: (1) evaluate recent operating experience with respect to the detection and sizing of circumferential indications; (2) develop a safety assessment justifying continued operation until the next scheduled steam generator tube inspections are performed; and (3) develop plans for the next inspections of steam generator tubes as they pertain to the detection of circumferential cracking. The error and additional information described above under "Description" do not change the basis for the actions required in the GL. Therefore, issuance of an IN is an appropriate regulatory action to correct the error and provide licensees with the additional information.

Resolution: IN 95-40, "Supplemental Information to Generic Letter 95-03, `Circumferential Cracking of Steam Generator Tubes," was issued on 09/20/95.

Completion Date: 09/20/95

GCCA-0029: REACTOR COOLANT PUMP TURNING VANE BOLT LOCKING DEVICE FAILURE

TAC No.: M92027 **Contact:** E.N. Fields

Description: On 06/03/94, the licensee for Seabrook conducted an underwater examination of reactor vessel internals. Foreign material was found on the reactor vessel internals lower core plate. In a subsequent video inspection, a bolt was found on the bottom of the reactor vessel and two bolt-locking devices were found on the lower core plate. One locking device was intact and the other was deformed and had portions missing. The licensee identified the bolt and locking devices as a cap screw and locking cups that are used in the RCPs to attach and secure the turning vane diffuser to the thermal barrier flange.

The degradation of the locking device and the release of the bolt was evaluated by the licensee with assistance from **W**. The root cause of the release of the turning vane cap screw and locking cups was attributed to the original design not adequately considering the affects of flow-induced vibration on the locking cup and the turning vane cap screw. The licensee postulated that flow-induced vibration caused the locking cups to erode and release from the turning vane. The cap screw subsequently backed out as a result of the loss of the pre-load torque and the effects of vibration and gravity.

Originating Document: LER No. 94-010-01.

Regulatory Assessment: From a safety perspective, the failure of a locking cup and the resultant release of a turning vane cap screw, in the worse case, could result in fuel damage and/or subsequent fuel failure. Licensees would likely be alerted to these failures by the loose parts monitoring system. However, the point at which loose parts noise activity would force a reactor shutdown is largely site-specific. If fuel damage resulted from the impact of loose parts, the licensee would be alerted by increased RCS activity, either by radiation monitors or required RCS sampling. Maximum allowable RCS activity is controlled by TS. Any increase beyond TS limits would force a reactor shutdown to evaluate the source of the activity.

Resolution: IN 95-43, "Failure of the Bolt-Locking Device on the Reactor Coolant Pump Turning Vane," was issued on 09/28/95.

Completion Date: 09/28/95

GCCA-0030: MAIN STEAM ISOLATION VALVE FAILURE DUE TO PILOT VALVE MALFUNCTION

TAC No.: M92028 **Contact:** D.L. Skeen

Description: Sticking solenoid pilot valves (Automatic Switch Company [ASCO] model NP8323) caused two MSIVs to fail to close at LaSalle-2; additionally, one MSIV at LaSalle-1 closed after a 15-second delay. These events occurred in February and June 1995, respectively. The root cause was determined to be contamination of internal piece parts (the core assembly and plug nut) that caused them to stick together. The contamination is believed to be a combination of lubricant (Nyogel 775A) used by the manufacturer during assembly of the NP8323 SOVs and thread sealant used when connecting the air lines to the pneumatic actuator, of which the NP8323 valve is a part.

The manufacturer discontinued production of the model NP8323 SOV in 1990 due to problems with foreign material entering the valves and causing either inhibited movement or degradation of the ethylene propylene elastomers. The manufacturer believed that the inhibited movement was the result of lubricant or thread sealant from the MSIV pneumatic actuator assembly and the degradation of the elastomers was the result of ester oils used in the air compressors of the instrument air systems at nuclear power plants.

Even though ASCO believed that field conditions rather than design defects were responsible for the

problems seen in the NP8323 valves through 1990, they notified customers at that time that the NP8323 SOVs were being discontinued and should be replaced in a timely manner. Some customers replaced the SOVs, but other customers who had not experienced any problems ordered more of the NP8323 valves before production was halted. Since it was not clear how many licensees may still be using these SOVs, an IN was issued by the NRC to alert licensees to the recent problems experienced by LaSalle.

Originating Documents: NRC Region III Morning Report 3-95-0019, dated 02/22/95; 10 CFR 50.72, Event Notification 28926, dated 06/11/95.

Regulatory Assessment: The MSIVs function to limit the release of radioactive materials to the environment or limit reactor coolant inventory loss given a steam line break. There are two MSIVs in series for each main steam

line either one of which is capable of isolating the line if needed. The scenario of concern would be a steam line break in a line where both MSIVs failed to close because of sticking pilot valves.

Resolution: IN 95-53, "Failures of Main Steam Isolation Valves as a Result of Sticking Solenoid Pilot Valves," was issued on 12/01/95.

Completion Date: 12/01/95

GCCA-0031: POTENTIAL CABLE DAMAGE FROM EXCESS SIDE WALL PRESSURE

TAC No.: M92215 **Contact:** J.L. Birmingham

Description: This proposed IN discussed the concern that some licensees had not considered cable sidewall bearing pressure when calculating allowable pull tension. The proposed IN also discussed instances that occurred during the past five years in which cable sidewall bearing pressure requirements had been exceeded or had not been documented in procedures used for cable pulls. However, these instances did not result in identified cable damage.

Originating Document: The IN Authorization Form (from R.L. Spessard to A.E. Chaffee) signed on 3/27/95 proposed an IN on failure to consider cable sidewall bearing pressure during cable pulls.

Regulatory Assessment: During review of the proposed notice, a thorough search of NRC and industry data sources for similar occurrences was made. This search did not find any additional recent examples of failure to consider cable sidewall bearing pressure. The NRC had addressed an extensive failure to consider cable

sidewall bearing pressure that occurred at Watts Bar-2. As a result of that occurrence, the NRC issued IN 92-01, "Cable Damage Caused by Inadequate Cable Installation Procedures and Controls." Because the concern does not appear to be a current problem and because the concern was previously addressed by the NRC in a generic communication, issuance of the proposed notice is not necessary at this time.

Resolution: The need for the proposed IN was discussed at a meeting of the Events Assessment Panel on 08/01/95 and the proposed notice was determined not to be necessary and was cancelled. The determination was based on the concern being primarily plant-specific.

Completion Date: 08/01/95

GCCA-0032: EVALUATE MISSILES FROM MIRROR INSULATION DURING HIGH ENERGY PIPE BREAKS

TAC No.: M92216 **Contact:** J.L. Birmingham

Description: During the NRC staff's evaluation of actions to prevent clogging of suppression pool strainers, a concern was identified that, in the event of a LOCA, reflective metallic insulation (RMI) debris may form missiles and damage safety system components such as control or power cabling.

Originating Document: Chapter 5 of the Draft Report, "Knowledge Base for Emergency Core Cooling System Recirculation Reliability," being prepared for the Organization for Economic Cooperation and Development/ Nuclear Energy Agency.

Regulatory Assessment: Initial assessment of this concern determined that the metallic foil in mirror insulation could potentially damage control or power cabling during a LOCA and affect the operability of safety systems inside containment. However, the concern was mitigated because redundant safety systems are typically isolated by distance or missile shielding and because critical components for mitigating an accident, such as ECCS pumps, are not located inside the drywell. Further, cables inside

the drywell are generally in conduits or cable trays which are of a durable construction that would make it difficult for the debris to damage cabling.

After the staff's initial assessment, two tests have been conducted which appear to substantiate the assessment. The first test was sponsored by the NRC and was conducted at the Siemens facility in Karlstein, Germany. The second test was performed by TRANSCO Products Inc., an RMI vendor. The results of both tests showed that little significant damage was done to the cable jacket material by the metallic foil in the RMI and that the metallic foil did not damage the cable itself.

In the first test, some of the cabling was severed. This is believed to have been caused by large chunks of the RMI panels impacting on the cables. The staff noted that the cabling was hung in a manner atypical of domestic BWRs in that the cabling was not installed in cable trays or conduits and was tightly hung with little slack in

the cable which did not allow the cable to give on impact with the larger RMI debris. The larger pieces of RMI

debris were readily stopped by intervening structures and were found to be wrapped around them. The staff has concluded that cable trays and conduits, as well as containment structures and piping, are likely to provide additional protection from large RMI debris.

Resolution: Based on evaluation of the test results, the staff recommended that no further action be taken on this issue at this time. In addition to the low safety significance, the staff found that the issue is very plant-specific and break location-specific and is therefore best addressed by the individual licensees. The staff intends to include a discussion on this issue in the final bulletin addressing the strainer issue so that licensees who have

or intend to install RMI will be advised of the potential problem and will account for it in their final resolution. The staff documented this resolution in a memorandum from R.B. Elliott to C. Berlinger on 11/07/95. The memorandum has additional details on the safety assessment and testing.

Completion Date: 11/07/95

GCCA-0033: RESULTS OF RECENT NRC SPONSORED FLAME SPREAD AND FIRE ENDURANCE TESTING

TAC No.: M92406 **Contact:** T.J. Carter

Description: Flame spread tests involving Thermo-Lag 330-1 fire barrier panels were conducted 01/12/95. The results have not been provided to licensees for determination of applicability to their facilities. Previous concerns had been raised regarding performance of Thermo-Lag as a fire barrier material.

Originating Document: None.

Regulatory Assessment: Licensees have taken compensatory measures as a result of concern about the Thermo-Lag performance. Therefore, there is time to conduct tests and provide licensees with information as it becomes available.

Resolution: IN 95-32, "Thermo-Lag 330-1 Flame Spread Test Results," was issued on 08/10/95 to provide results of the flame spread tests.

Completion Date: 08/10/95

GCCA-0034: COMMISSION DECISION ON THE RESOLUTION OF GENERIC ISSUE 23, "REACTOR COOLANT PUMP SEAL FAILURE"

TAC No.: M92408 **Contact:** T. Koshy

Description: On 04/19/91, the NRC published Federal Register Notice 56 FR 16130 requesting comments on the then-current understandings, findings, and potential recommendations regarding GI-23, together with a

draft Regulatory Guide, DG-1008, "Reactor Coolant Pump Seals." On 05/02/91, NRC issued GL 91-07, "GI-23,

'Reactor Coolant Pump Seal Failures' and Its Possible Effect on Station Blackout," which stated that preliminary results of NRC studies suggested that RCP seal leak rates could be substantially higher than those assumed

in the coping analyses for implementation of the station blackout (SBO) issue. The GL reminded licensees that higher seal leak rates could affect licensee analyses and actions addressing conformance to the SBO rule.

Staff studies and analyses concerning RCP seal leakage are documented in Appendices A and B to NUREG/ CR-5167, "Cost/Benefit Analysis for Generic Issue 23: Reactor Coolant Pump Seal Failures," April 1991, which contains the NRC model for RCP seal failure. The report identifies several modes of RCP seal leakage which may be in excess of that assumed in licensee coping analyses for implementing the requirements of 10 CFR 50.63, the SBO rule.

The Commission considered the proposed rulemaking as a method to resolve GI-23. In SECY-94-225, dated 08/26/94, a draft rule was proposed for public comment that would resolve GI-23. On 03/31/95, the Commission voted against publication of the proposed rule that would have resolved GI-23. The Commission concluded

that the proposed rule did not provide sufficient gain in safety to justify its issuance. The Commission was also concerned that inaccuracies in the NRC seal leakage evaluation model may exist.

Originating Document: SRM dated 03/31/95.

Regulatory Assessment: The NRC staff had conducted extensive study on the vulnerability of RCP seal failures and considered issuance of generic requirements to address this issue. Since this issue was identified

as a GI, the industry has addressed the problem. Improvements in RCP seal designs and the exchange of coping analyses information, etc., are the outcome of these efforts.

Even though a new rule was not issued, the staff efforts so far has resulted in increased sensitivity to this issue and has resulted in a documented coping analysis through the SBO rule at every station. The staff continues to evaluate if any further action is needed for the disposition of this issue.

Resolution: IN 95-42, "Commission Decision on the Resolution of Generic Issue 23, `Reactor Coolant Pump Seal Failure,'" was issued on 09/22/95.

Completion Date: 09/30/95

GCCA-0035: VOLTAGE-BASED INTERIM REPAIR CRITERIA FOR STEAM GENERATOR TUBES

TAC No.: M92578 **Contact:** J.W. Shapaker

Description: PWR facility licensees may, on a voluntary basis, request a license amendment to the plant TS to implement alternate SG tube repair criteria applicable specifically to outside diameter stress corrosion cracking (ODSCC) at the tube-to-tube support plate intersections in **W**-designed SGs having drilled-hole tube support plates (TSP) and Alloy 600 tubing. Guidance is offered on implementing the alternate (voltage-based) repair criteria. By approving the use of specific voltage-based repair criteria as an acceptable measure for dealing with ODSCC tube degradation, it is intended to provide relief while maintaining an acceptable level of safety for licensees having SGs experiencing this particular degradation mechanism until a longer-term resolution to the issue of SG degradation is pursued through rulemaking.

Originating Document: None.

Regulatory Assessment: Design of the RCPB for purposes of structural and leakage integrity is a requirement under 10 CFR 50, Appendix A, GDC for Nuclear Power Plants. Specific requirements governing the maintenance of SG tube integrity are in plant TS and in Section XI of the ASME Boiler and Pressure Vessel Code (ASME Code). These include requirements for periodic ISI of the tubing, flaw acceptance criteria (i.e., repair limits for plugging or sleeving), and primary-to-secondary leakage limits. These requirements, coupled with the broad scope of plant operational and maintenance programs, have formed the basis for ensuring adequate SG tube integrity.

The traditional strategy for achieving the objectives of the GDC regarding SG tube integrity has been to establish a minimum wall thickness requirement in accordance with the structural criteria of RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." A further assumption of uniform thinning results in the development of a repair limit (40% depth-based repair limit is typically incorporated into plant TS) that is conservative for all flaw types.

The NRC has approved for use, on a voluntary basis, alternate (voltage-based) repair criteria that are only applicable to predominantly axially-oriented ODSCC indications located at the tube-to-TSP intersections in **W**-designed SGs with Alloy 600 tubing. (More explicit guidance on applicability constraints, voltage-based repair limits, and implementation actions are provided in GL 95-05.) The voltage-based repair criteria ensure structural and leakage integrity for all postulated design basis events. The structural criteria are intended to ensure that indications subjected to the voltage repair limits will be able to withstand pressure loadings consistent with the criteria of RG 1.121. The leakage criteria ensure that for degradation subjected to the voltage repair criteria, induced leakage under the worst-case main steam line break conditions calculated using licensing basis assumptions will not result in offsite and control room dose releases that exceed the applicable limits of 10 CFR 100 and GDC 19.

Resolution: GL 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," was issued on 08/03/95.

Completion Date: 08/16/95

**GCCA-0036: FAILURE OF AUTOMATIC VENTILATION SYSTEM OPERATION
FOLLOWING A LOSS OF OFFSITE POWER**

TAC No.: M92596 **Contact:** J.R. Tappert

Description: At Waterford-3, the licensee rendered several ESF ventilation systems inoperable under certain conditions because of an engineering oversight in a design change installed between October 1992 and April 1993.

Waterford has four different ESF ventilation systems that are used to filter radioactive material from the air during an accident. These include: (1) the shield building ventilation system, which filters the air being removed from the shield building annulus and maintains a partial vacuum inside the annulus; (2) the controlled ventilation

area system, which filters air from rooms containing the emergency core cooling and containment heat removal systems in the reactor auxiliary building; (3) the control room air system (CRACS), which filters the air going into the control room; and (4) the fuel handling building ventilation system, which filters the air being removed from the fuel handling building. Each ventilation system contains two independent trains. Each train has a demister, electric heating coil, pre-filter, HEPA pre-filter, charcoal adsorber, HEPA after-filter, and 100%-capacity exhaust fan.

Automatic temperature controllers monitor the air temperature and cycle the heaters to maintain the desired air temperature. To improve the system performance and reduce system trips, the licensee developed a design change to replace the heater controllers with more accurate controllers. Because of an engineering oversight, the design change caused the control circuits for the ventilation system to trip the heaters after the ventilation systems lost power. Specifically, when de-energized, the temperature controllers defaulted to the high temperature trip setpoint. Upon re-energization, the new temperature controllers responded more slowly

than the 0.5 second allowed to indicate that the temperature had dropped below the high temperature condition. Following a Loss of Offsite Power (LOOP) in conjunction with a safety injection actuation signal, the 0.5-

second time-delay relay would time out and trip the ventilation heaters. With the heaters tripped, the differential temperature would decrease across the heaters and eventually trip the ventilation fan.

Originating Document: Event Notification 27206, dated 05/03/94.

Regulatory Assessment: The controller design change created a common failure mode. Three of the four ESF ventilation systems were inoperable under certain conditions from the time the design change was installed in 1993 until May 1994. The fourth system, CRACS, was operable only because timer relays had been inadvertently left out of the preventive maintenance program and had drifted out of specification. If the fans

in these ventilation systems trip, then offsite and control room radiological dose during a DBA could increase above the calculated value. Without the ventilation fans, the annulus pressure in the shield building would increase above the atmospheric pressure. Leakage from containment would be unfiltered and would release directly to the atmosphere. This direct release could have significant impact on the accident doses. However, the failure mode is limited to certain failure sequences and the heater controllers can be reset locally. The licensee calculated that operator exposure to reset the heaters would be less than 1.2 rem if the LOOP was concurrent with the LOCA and less than 7 rem if the LOOP occurred, at a worst case, 4 days after the LOCA. An IN was proposed to notify other licensees of this event but was subsequently cancelled.

Resolution: After management review, the generic communication was cancelled based on the B. Grimes memorandum to NRR Division Directors, "Advance Notification of the Division of Project Support's Intent to No Longer Support Development and Issuance of Selected Generic Letters and Information Notices," dated 06/10/95. The proposed IN failed to meet the threshold for regulatory action based on the following criteria: (1) length of development; (2) number of potentially affected plants; and (3) safety significance.

Completion Date: 07/21/95

GCCA-0037: FAILURE TO TEST SWING BUSES DURING INTEGRATED EMERGENCY DIESEL GENERATOR SURVEILLANCE

TAC No.: M92597 **Contact:** T. Koshy

Description: On 09/07/94, Waterford-3 reported to the NRC that one train of safety-related equipment was inoperable since the TS-required surveillances were not performed to ensure operational readiness. The Waterford station has a third train of safety-related equipment that can be used to substitute the function of the "A" or "B" train of equipment. In the past, surveillance required by the TS was not conducted on this spare train.

Originating Document: 09/07/94 Phone call to the NRC for enforcement discretion.

Regulatory Assessment: When the NRC was made aware of this scenario, the staff reviewed and approved the licensee's bases in continuing the operation in light of the historic performance of these components, even though a prescribed surveillance record was unavailable.

Initially, an IN was considered to promulgate the need for testing spare components that are in service to ensure its operational readiness. Later, it was recognized that the theme of this IN was contained in another recently issued IN 95-15, "Inadequate Logic Testing of Safety-Related Circuits." Therefore, an additional IN was considered unnecessary.

Resolution: None required.

Completion Date: 07/20/95

GCCA-0038: LIGHTNING DISSIPATION SYSTEMS

TAC No.: M92599 **Contact:** N.K. Hunemuller

Description: Lack of effectiveness of lightning dissipation systems in protecting plant electronics.

Originating Document: None.

Regulatory Assessment: This concern was not safety-significant and was cancelled by the originator prior to any development.

Resolution: After management review, the generic communication was cancelled based on B.W. Sheron's 07/21/95 memorandum to B.K. Grimes, "Response to the Division of Project Support's Request for Review of Generic Compliance and Communications Activities (GCCA)." The proposed generic communication failed to meet the threshold for regulatory action based on its safety significance.

Completion Date: 07/21/95

GCCA-0039: SWITCHGEAR FIRE AND PARTIAL LOSS OF OFFSITE POWER

TAC No.: M92600 **Contact:** E.J. Benner

Description: On 06/10/95 at Waterford-3, a generator trip occurred in response to failure of a lightning arrester on a remote offsite substation transformer. The generator trip resulted in a fast transfer activation. A fire and electrical fault occurred on the 4.16kV A2 bus normal power supply breaker. The 6.9kV A1 bus alternate supply breaker failed to close resulting in a loss of power to RCPs 1A and 2A. This circumstance resulted in a reactor trip and a loss of offsite power to the 4.16kV non-safety-related A2 bus and the associated 4.16kV safety-related A3 bus. An auxiliary operator informed the control room of heavy smoke within the turbine generator building.

At that time, the shift supervisor did not activate the plant fire alarm or dispatch the fire brigade, but directed two auxiliary operators to don protective gear and investigate whether a fire existed. The operators reported seeing flames above the A2 switchgear and the shift supervisor activated the fire brigade. Operators requested assistance from the local offsite fire department. The fire brigade was unable to suppress the fire using portable fire extinguishers. The offsite fire department arrived on the scene and extinguished the fire with water after

the A2 bus was deenergized. A reflash occurred which had to be put out with water a second time. During the cooldown transition from Mode 4 to Mode 5, operators discovered that the isolation valves for both trains of shutdown cooling did not operate properly. The plant cooldown to Mode 5 was delayed approximately 38 hours while these valves were repaired.

Originating Document: Event Notification 28923, dated 06/10/95.

Regulatory Assessment: The NRC conducted an augmented inspection team (AIT) inspection to

determine the causes, conditions, and circumstances relevant to this event. The results of this AIT inspection are documented in NRC Inspection Report 50-382/95-15, dated 07/07/95. The AIT identified three primary concerns: fire protection, fast bus transfer design, and shutdown cooling valve inoperability. Evaluation of these three concerns by NRR revealed that no immediate generic safety issue existed and that dissemination of the information to the industry would be an appropriate technical resolution.

Resolution: IN 95-33, "Switchgear Fire and Partial Loss of Offsite Power at Waterford Generating Station, Unit 3," was issued on 08/23/95.

Completion Date: 08/23/95

GCCA-0040: SPENT FUEL TRANSFER CANAL SHIELDING DEFICIENCY AT BOILING WATER REACTOR

TAC No.: M92876 **Contact:** T.A. Greene

Description: In November 1994, contractors at Hatch were conducting underwater operations in the Unit 1 spent fuel pool - cutting coupons out of spent control rod blades containing the upper guide roller bearings. The highly activated stellite bearings (some measured as high as 160 sievert [16,000 rem] per hour at 30 centimeters [12 inches] under water) were being collected adjacent to the Unit 1 work area. Periodically, the coupons containing the upper guide roller bearings were transferred from the collection bucket to a cask liner in the shipping cask storage area in the Unit 2 spent fuel. When the workers could not find the tool to open the

liner, they decided to transfer the coupons in the collection bucket (about half full, with 160 coupons) into another bucket for temporary storage, so that the cutting process could continue. To facilitate the task, the off-load was performed in the transfer canal since the canal was much shallower than the fuel pools. During the transfer, some of the coupons fell to the bottom of the transfer canal. Since the transfer canal was designed and routinely used as a transit area for highly activated material, including spent fuel, the workers were not concerned about dropping the coupons and saw no need to notify the unit shift foreman or the control room of the incident. They recovered the coupons from the bottom of the canal and placed them in the storage bucket resting on the canal floor.

About 30 minutes after the coupons were dropped, a plant operator was walking through the Unit 1 hallway directly under the transfer canal when his digital alarm dosimeter alarmed on high dose rate. The plant operator left the area promptly and notified a health physics supervisor. The licensee measured radiation levels of up to 1 sievert [100 rem] per hour on contact with the hallway ceiling directly below the bottom of the canal and 0.05 to

0.1 sievert [5 to 10 rem] per hour in the general area of the hallway.

Originating Documents: Inspection Reports 50-321/95-01 and 50-366/95-01 [Accession No. 9502140081].

Regulatory Assessment: The plant equipment operator received a dose equivalent of about 0.1 msievert [10 mrem] from the event, which is below the regulatory limit. The licensee has always required that all personnel entering the radiologically-controlled area be issued dosimeters. Before alarm dosimeters were required, all workers entering the radiologically-controlled area were issued personnel dosimeters (non-alarm), so any doses to workers from this shielding deficiency would have been detected and accounted for as part of the routine dosimetry program.

Many licensees perform spent fuel pool modifications and major cleanup activities involving handling and moving large quantities of highly activated materials, including spent fuel. In general, the industry has significantly improved its awareness of and controls for potentially high and very high radiation areas caused by operational mishaps (e.g., dropping a spent fuel bundle in the transfer canal directly over the upper drywell in a BWR, and the hazards of withdrawn in-core thimbles under the reactor vessel at PWRs). Initiatives taken by licensees should help prevent unexpected, uncontrolled worker exposures with the potential for exceeding the regulatory limits. A thorough pre-job evaluation of activities involving highly activated (or potentially highly activated) components can help identify challenges to existing plant shielding.

Resolution: At Hatch, the licensee instituted procedural control to prohibit the use of the transfer canal until doors had been installed in order to exclude worker access to the affected hallway. Before the transfer gates are allowed to be lifted (allowing use of the transfer canal), the doors at each end of the hallway under the canal are locked and access to the hallway is controlled as a very high radiation area. IN 95-56, "Shielding Deficiency in Spent Fuel Transfer Canal at a Boiling-Water Reactor," was issued on

12/12/95 to alert the industry.

Completion Date: 12/11/95

GCCA-0041: CHANGES IN THE OPERATOR LICENSING PROGRAM

TAC No.: M92877 **Contact:** J.W. Shapaker

Description: The NRC intends to change the operator licensing process to give facility licensees the option of preparing written examinations and operating tests for operator and senior operator license applicants. The NRC will review and approve licensee-proposed examinations and tests and will independently conduct the operating tests. Facility licensees will only conduct the written examinations. The NRC will review the graded written examinations, grade each applicant's operating test performance, make the final pass or fail decisions,

and issue licenses, as appropriate. From October 1995 to March 1996, a voluntary pilot program to evaluate and refine the new examination development process will be conducted.

Originating Documents: NUREG-1021, Revision 7, "Operator Licensing Examiner Standards"; NUREG/BR-0122, "Examiners' Handbook for Developing Operator Licensing Written Examinations"; GL 95-06, "Changes in the Operator Licensing Program."

Regulatory Assessment: 10 CFR 55, "Operators' Licenses," establishes NRC procedures and criteria for issuing licenses to operators and senior operators. It does not, however, define a specific process for conducting licensing examinations. Guidance in this area is given in NUREG-1021, "Operator Licensing Examiner Standards," which includes the procedures that NRC staff examiners and NRC-certified contract examiners

use to prepare and conduct both the written and operating portions of the licensing examinations. The role of power reactor facility licensees had been limited to reviewing and validating the NRC-prepared examinations before they are given, and to providing administrative and logistical support to the NRC and contract examiners while the examinations are in progress. Over the past 10 years, power reactor licensees have placed increased emphasis on establishing accredited training programs and, as a result, improvements in operator training

and performance have been evident in the initial operator licensing process. This fact, in conjunction with a continuing effort on the part of the NRC to streamline the functions of the Federal Government consistent with White House initiatives and to accommodate anticipated resource reductions, motivated the NRC to reconsider its approach to conducting the initial operator licensing examination program. The NRC intends to change

the guidance in NUREG-1021 to permit facility licensees greater participation in the initial operator licensing program.

Resolution: GL 95-06, "Changes in the Operator Licensing Program," was issued on 08/15/95.

Completion Date: 08/16/95

GCCA-0042: UNPLANNED, UNMONITORED RELEASE OF RADIOACTIVITY FROM THE EXHAUST VENTILATION SYSTEM OF A BWR

TAC No.: M92935 **Contact:** C.V. Hodge

Description: On 04/05/95, an unplanned, unmonitored release of high specific activity-mixed radioactive corrosion products was released from the south plant vent of Hope Creek. The licensee believed that no release had occurred because of a lack of unusual indications. Large portions of the protected area were contaminated, including onsite vehicles and buildings, and contamination levels accessible to personnel was about 10^5

dpm/100cm². One vehicle, located downwind of the release point, left the site and was determined two days later to be contaminated. The radioactive contamination came from a small capacity evaporator for decontamination solution discharging directly to the south plant vent in the turbine building. The licensee believed, from lack of unusual indications, that no release had occurred. About 14 hours into the event, the licensee observed high contact dose rates from effluent ducts and highly contaminated liquid leaking from the ducts. The leaking liquid was attributed to a pre-existing condition.

Originating Document: Inspection Report 50-354/95-05.

Regulatory Assessment: Region I conducted a special inspection and found that: (1) no release limits were exceeded; (2) worst-case analyses indicated that the release had little radiological effect on the public and environment; (3) safe operation of the reactor was not affected; (4) the release did not

² NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.

significantly affect onsite personnel; and (5) the licensee did an excellent evaluation of the environmental effect of the release. However, Region I concluded that: (1) there were four apparent violations for escalated enforcement existed related to

(a) lack of adequate approved operating procedures for this evaporator (TS 6.8.1), (b) inadequate monitoring to enable detection of the release [10 CFR 20.1501(a)], (c) alarm setpoint changes not made in accordance with approved procedures (TS 6.8.1), and (d) workers not being informed of the release and onsite contamination once it was identified (10 CFR 19.12); (2) there was a particular concern about the inability of licensee personnel to readily determine that a release had occurred; and (3) there was a particular concern that licensee design reviews did not identify the potential for an unmonitored release, despite previous NRC-published information

on this subject such as (a) IN 91-40, "Contamination of Nonradioactive System and Resulting Possibility for Unmonitored, Uncontrolled Release to the Environment," dated 06/19/91, (b) Circular 80-18, "10 CFR 50.59 Safety Evaluations for Changes to Radioactive Waste Treatment Systems," dated 08/22/80, and (c) Bulletin 80-10, "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to the Environment," dated 05/06/80. This particular event represents minimal safety significance; however, its potential safety significance and potential generic implication support the Region I recommendation, endorsed by TERB/DOTS staff, to issue the proposed IN.

Resolution: IN 95-46, "Unplanned, Undetected Release of Radioactivity from the Exhaust Ventilation System of a Boiling Water Reactor," was issued on 10/06/95.

Completion Date: 10/06/95

GCCA-0043: SUSCEPTIBILITY OF LOW PRESSURE COOLANT AND CORE SPRAY INJECTION VALVES TO PRESSURE LOCKING

TAC No.: M92960 **Contact:** T.A. Greene

Description: From May to July 1995, Georgia Power Company, the licensee for Hatch-1 & 2, experienced several valve failures in both units during testing of its in-board LPCI 24-inch flexible gate valve. During the investigation of the root causes for these failures and in responding to the NRC staff's inquiries, the licensee realized that, because of back-leakage of the downstream check valves, reactor system pressure could cause the LPCI valves and the core spray injection valves to be susceptible to pressure locking during an accident.

Originating Document: Inspection Report 50-321/95-17, dated 08/24/95.

Regulatory Assessment: The NRC staff and the nuclear industry have been aware of pressure-locking problems in gate valves for many years. The industry has issued several event reports describing failures of safety-related gate valves to operate because of pressure locking. Several generic industry communications have given guidance for identifying susceptible valves and for taking appropriate preventive and corrective measures. Also, the NRC staff has provided information on pressure locking of gate valves to the industry and has discussed the safety significance of the potential for pressure locking of gate valves at public meetings.

Resolution: The generic concern was resolved by the issuance of IN 95-30, "Susceptibility of Low-Pressure Coolant Injection and Core Spray Injection Valves to Pressure Locking," and GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves."

Completion Date: 08/03/95

GCCA-0044: POTENTIALLY NONCONFORMING FASTENERS SUPPLIED BY A&G ENGINEERING II, INC.

TAC No.: M93226 **Contact:** J.R. Tappert

Description: IN 95-12, "Potentially Nonconforming Fasteners Supplied by A&G Engineering II, Inc.," was issued on 02/21/95 to alert licensees to potentially non-conforming fasteners supplied from A&G Engineering II, Inc.

Additional information concerning de-carburization and the manufacturer of the fasteners was reviewed after IN 95-12 was issued. A supplement was proposed to relay this additional information to licensees.

Originating Document: OI Report 5-93-016R re A&G Engineering II, Inc.

Regulatory Assessment: PECO found that 143 out of 150 heavy hex nuts supplied by A&G and an intermediary vendor HUB, Inc. had carbon contents significantly different from that reported on the Certified Material Test Report (CMTR). The nuts were certified by A&G to be traceable to Hamanaka Nut Mgn. Co. Ltd. The NRC has reviewed additional information since the issuance of IN 95-12. Further analysis of the rejected nuts from PECO found that, when the surface material was removed, the remaining metal had the proper carbon content. This would indicate that surface decarburization was primarily responsible for the previously identified low carbon values. However, it is not clear what effect the observed surface decarburization has on the thread integrity of the nuts. The NRC has also been contacted by the Hamanaka Nut Manufacturing Co., Ltd.

Hamanaka believes that the rejected nuts were fraudulently labeled with their trademark. Hamanaka hot-forms their trademark (raised), whereas the 143 rejected nuts had a stamped Hamanaka marking.

This concern is generic because A&G had supplied a significant amount of safety-related threaded fasteners to the nuclear power industry. The proposed IN supplement updates the industry on the recent information concerning surface decarburization and the Hamanaka trademark issues. The actual safety significance of the surface decarburization has not been ascertained. The larger issue of substandard/nonconforming materials being supplied to utilities is being addressed by the Special Inspection Branch (TSIB). They are performing a

number of inspections of vendors and utilities to determine the scope of the problem. Additionally, RES has been requested to determine what level of sampling is required for commercial grade dedication. TSIB proposes to issue a generic communication in the near future documenting their findings.

Resolution: Supplement 1 to IN 95-12 was issued on 10/05/95 to alert licensees to relevant plant operating experiences.

Completion Date: 10/05/95

GCCA-0045: CURRENT FIRE ENDURANCE TEST RESULTS FOR 3M INTERAM RACEWAY FIRE BARRIER SYSTEMS

TAC No.: M93295 **Contact:** T.J. Carter

Description: A number of full-scale fire endurance tests were sponsored by Peak Seals Corporation. The results of the fire endurance tests for electrical raceway fire barrier systems constructed from 3M Company Interam fire barrier materials have not been provided to licensees.

Originating Document: None.

Regulatory Assessment: The staff was informed about the test program and witnessed the tests. To provide information to licensees as soon as possible, the staff proposed that an IN be issued to present their observations. The staff has not reviewed the test reports.

Resolution: IN 95-52, "Fire Endurance Test Results for Electrical Raceway Fire Barrier Systems Constructed from 3M Company Interam Fire Barrier Materials," was issued on 11/14/95 to inform licensees of fire endurance test results.

Completion Date: 11/14/95

GCCA-0046: POTENTIAL FOR DATA COLLECTION EQUIPMENT TO AFFECT PROTECTION SYSTEM PERFORMANCE

TAC No.: M93359 **Contact:** E.M. McKenna

Description: Quad Cities-2 found that a portable computerized data acquisition system (DAS), connected to instrumentation circuits to collect data, was affecting the signals being monitored. Two different problems occurred:

(1) In December 1994, the licensee found that a RPS setpoint on low water level was reading out of tolerance with the DAS connected to the circuit, but de-energized. Further evaluation found that in this configuration, the DAS was a low internal impedance path in the circuit, allowing more current to be drawn and decreasing the signal. This problem is similar to an event at Fermi-2 (that occurred in February 1995) that was the subject of IN 95-13.

(2) In July 1995, the licensee re-installed the DAS to collect further data. Instructions and precautions were taken for the system not to be de-energized while connected (to avoid the above problem). However, two unexpected interactions of the energized DAS with the circuits being monitored occurred. First, there was a change in recirculation pump speed (in manual control) with no change demanded by the operator. Later, one water level channel was indicating 8 inches low (and there were also changes in total indicated core flow). The DAS was connected to the circuits being monitored by a ribbon cable that was unshielded from a terminal block external to the DAS to a circuit board within the DAS. The signals being processed are DC signals. Pre-installation testing had shown that stray AC voltage signals (present because of AC power supply to circuits) were present in the circuit. This AC voltage normally did not affect proper functioning of the circuits. When the DAS was in use, investigation revealed that electromagnetic interference across circuits within the ribbon cable caused feedback from one circuit into another. This feedback would result in a signal that would add or subtract from the desired signal being processed and resulted in changes in both control (recirculation pump) and indication.

Originating Document: Morning Report 3-95-0130, dated 08/07/95.

Regulatory Assessment: The DAS unit was considered by the licensee to be non-intrusive, so detailed testing of the possible effects of the unit on parameters being monitored was not performed. Redundant

channels

of instrumentation were simultaneously connected to the DAS, compromising the electrical separation of independent channels that is required. Most, but not all, of the channels affected at Quad Cities were non-safety related circuits. Only small changes in signals resulted, which were detected by plant personnel, and redundant channels were not simultaneously affected. An IN had been issued on the Fermi event (where the DAS was

de-energized). To reinforce the need to consider possible interactions of a DAS (especially since it may be connected to redundant channels), a supplement to the IN was issued.

Resolution: IN 95-13, Supplement 1, "Potential for Data Collection Equipment to Affect Protection System Performance," was issued on 11/22/95 to alert licensees to the relevant plant operating experience, but does not require any licensee action.

Completion Date: 11/22/95

GCCA-0047: BORAFLEX DEGRADATION IN SPENT FUEL POOL STORAGE RACKS

TAC No.: M93373 **Contact:** P.C. Wen

Description: At South Texas Project (STP), blackness (neutron absorption) testing was performed during August 1994 on selected Unit 1 spent fuel pool storage racks to determine the condition of the Boraflex. The test results indicated that the Boraflex had significantly degraded due to dissolution of the Boraflex in the spent fuel pool environment.

Originating Documents: Event Notification 28277 and LER 50-498/95-002.

Regulatory Assessment: On the basis of test and surveillance information from plants that have detected areas of Boraflex degradation, no safety concern exists that warrants immediate action. Boraflex dissolution appears to be a gradual and localized effect forewarned by relatively high silica levels in the pool water. Because of the safety margin present in spent fuel storage pools, compliance with the required sub-criticality margin

(or conformance with the same margin to which licensees have committed in their updated FSARs) can be expected to be maintained during the initial stage of Boraflex degradation. This safety margin is due to the 5% sub-criticality margin assumed in the analysis, the generally lower reactivity of stored fuel than that assumed in the safety analysis, and, in the case of PWRs, the presence of borated water in the pool. Therefore, continued facility operation is justified.

Boraflex degradation is a generic concern. Since there are over 50 pools that contain Boraflex and in light of the problems that were discovered at Palisades (IN 93-70), STP, and most recently at Fort Calhoun, this Boraflex concern may be a widespread problem.

The staff proposed that an IN be issued to inform licensees of this concern and that a GL be developed to require appropriate actions. TAC No. M93373 was initiated for the development of the IN.

Resolution: IN 95-38, "Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks," was issued on 09/08/95. A GL on this topic is being developed by SRXB.

Completion Date: 09/08/95

GCCA-0048: RISK IMPACT OF MAINTENANCE DURING LOW POWER OPERATION AND SHUTDOWN

TAC No.: M93642 **Contact:** N.K. Hunemuller

Description: The details of a study were published which evaluated the risk impact of LCOs at low power and shutdown in the current TS for Grand Gulf. A probabilistic model was developed for each of eight plant operational states. The study indicated that the increase in conditional CDF for taking a single train of standby

service water out of service during low power and the first few days of hot shutdown was comparable to that at full-power operation. During hot shutdown and cold shutdown, the study indicated that the increase in conditional CDF could exceed that at full-power operation.

Originating Document: NUREG/CR-6166, "Risk Impact of BWR Technical Specifications Requirements

During Shutdown," published October 1994.

Regulatory Assessment: On 05/23/95, the Generic Issues and Events Assessment Panel initially determined that there was a lack of sufficient data to support a conclusion of generic applicability on the specific issue of the impact of service water maintenance shortly after shutdown. However, the broad concern of low power and shutdown risk was generic. Although NUREG/CR-6166 was a public document, there was sufficient management interest to proceed with development of an IN to ensure that licensees were aware of the study. After further review, the Panel authorized development of an IN on 09/12/95.

Resolution: IN 95-57, "Risk Impact Study Regarding Maintenance During Low-Power Operation and Shutdown," was issued on 12/13/95 to inform licensees of an RES study on the effect of system or equipment maintenance on BWR low-power and shutdown CDF, but no licensee action was required.

Completion Date: 12/13/95

GCCA-0049: LEGAL ACTIONS AGAINST THERMAL SCIENCE, INC., MANUFACTURER OF THERMO-LAG

TAC No.: M93668 **Contact:** T.J. Carter

Description: An IN was previously issued to inform interested parties that there was an indictment pending against Thermal Science, Inc. A jury subsequently acquitted Thermal Science of all charges.

Originating Document: IN 94-86, "Legal Actions Against Thermal Science, Inc., Manufacturer of Thermo-Lag," issued 12/22/94.

Regulatory Assessment: This is not a safety issue. However, there is a need to clarify the record presented by the original IN.

Resolution: IN 94-86, Supplement 1, "Legal Actions Against Thermal Science, Inc., Manufacturer of Thermo-Lag," was issued on 11/15/95 to inform licensees that a jury acquitted Thermal Science, Inc. and its president of making false statements.

Completion Date: 11/15/95

GCCA-0050: TRANSIENT INVOLVING OPEN SAFETY RELIEF VALVE FOLLOWED BY COMPLICATIONS

TAC No.: M93705 **Contact:** T.J. Carter

Description: During routine operation of Limerick Unit 1 on 09/11/95, one 2-stage Target Rock SRV opened and stuck open causing a blowdown transient. This was the first time this particular type of SRV had unexpectedly opened; cause was attributed to excessive steam leakage through the pilot valve. Reason for the initial leakage is not known, but it is known that the stellite seat and disk had ultimately become badly eroded. Suppression pool cooling was ongoing to remove heat being added to the pool by the leaking steam. However, as a result

of the blowdown transient, additional pool cooling was required and a second pump was started. Erratic flow was observed subsequently on the initially running pump. This was interpreted as pump cavitation caused by debris clogging the suction strainer. Upon inspection, the suction strainer on the pump that exhibited erratic performance was found to be largely covered with a fibrous material and sludge.

Originating Document: Event Notification 29316, dated 09/11/95.

Regulatory Assessment: Sudden opening of an SRV is an initiating event that challenges safety systems and has been analyzed. However, excessive steam leakage past the pilot valve with the subsequent SRV opening is new and raises a generic concern since licensees cannot differentiate steam leaking through the main valve or the pilot valve. The previously observed accumulation of material on the suction strainers at several reactors has already led to a generic action plan which is still in progress. This event itself is of minor immediate concern since the cooling function was retained. However, the fact that the licensee had been operating for months

with leaking SRVs is indicative of a potential "work around" with unexpected consequences. Furthermore, the suppression pool at Limerick Unit 1 had never been cleaned even though an opportunity existed after the licensee had removed a substantial amount of material from the Unit 2 pool during its cleaning. This may demonstrate another lack of appreciation of the potential consequences. An IN was prepared rapidly to alert licensees to the potential precursor of spurious SRV opening and reiterate the concern about unclean

suppression pools. The degradation of strainer performance due to debris accumulation during normal operation may require additional generic action.

Resolution: IN 95-47, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," was issued on 10/04/95 to alert licensees to a failure of an SRV to remain closed during steady state operation and the ensuing complications involving blockage of a strainer located in the suppression pool.

Completion Date: 10/04/95

GCCA-0051: UNEXPECTED CLOGGING OF RHR PUMP STRAINER WHILE OPERATION IN SUPPRESSION POOL COOLING MODE

TAC No.: M93740 **Contact:** J.W. Shapaker

Description: On 09/11/95, Limerick-1 experienced a stuck-open SRV while operating at 100% power. Attempts to close the valve were unsuccessful and a manual reactor scram was initiated. The licensee initiated suppression pool cooling and subsequently experienced clogging of one RHR pump suction strainer.

Originating Document: Event Notification 29316, dated 09/11/95.

Regulatory Assessment: 10 CFR 50.46 requires that licensees design their ECCS so that the calculated cooling performance following a LOCA meets five criteria, one of which is to provide long-term cooling capability of sufficient duration following a successful system initiation so that the core temperature shall be maintained

at an acceptably low value, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core. The ECCS is designed to meet this criterion, assuming the worst single active failure and only partially obstructed flow through the strainer(s). The Limerick event demonstrated that inadequate suppression pool cleanliness can adversely impact ECCS performance and could prevent the ECCS from performing its safety function of long-term decay heat removal following a LOCA. As a result, BWR licensees were requested to take certain actions that, if required, would be compliance backfits under the terms of 10 CFR 50.109(a)(4)(i), i.e., would be necessary to ensure compliance with NRC rules and regulations. IN 95-47, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," was issued on 10/04/95 to alert licensees about the event and its consequences.

Resolution: Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode."

Completion Date: 11/26/95

GCCA-0052: UNEXPECTED OPENING OF AN SRV AND COMPLICATIONS INVOLVING SUPPRESSION POOL STRAINER BLOCKAGE

TAC No.: M93840 **Contact:** E.M. McKenna

Description: On 09/11/95, an SRV at Limerick-1 opened unexpectedly. An IN issued on 10/04/95 to alert licensees about the event required revision.

Originating Document: IN 95-47, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," dated 10/04/95.

Regulatory Assessment: This TAC was opened for issuance of a revision to IN 95-47. (See GCCA-0050 for further details on the initiating event and the IN.) In addition, Bulletin 95-02, "Unexpected Clogging of RHR Pump Strainer While Operating in Suppression Pool Cooling Mode," was issued on 10/17/95 (See TAC M93740).

The IN revision was proposed to clarify licensee planned actions when tail pipe temperature limits were reached and to note that the NRC plans to evaluate the potential generic implications of significant leakage through Target Rock 2-stage safety/relief valves, such as occurred at Limerick (this generic effort is being tracked under TAC No. M93841). Details on licensee proposed action plans are contained in a letter dated 10/06/95, from PECO Energy Company [Accession No. 9510120218].

Resolution: IN 95-47, Revision 1, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," was issued on 11/30/95.

Completion Date: 11/30/95

GCCA-0053: DECAY HEAT MANAGEMENT PRACTICES DURING REFUELING

TAC No.: M94087 **Contact:** D.L. Skeen

Description: Adequacy of decay heat management practices during refueling outages was called into question as a result of an NRC review of a design change at Millstone-1 (submitted to the staff on 07/28/95). The practice of unloading the entire core from the reactor vessel to the SFP during refueling

has become fairly common.

However, a full core off-load during refueling may potentially exceed the ability of the SFP cooling system to maintain SFP temperature within its design limit, if other means are not available to assist the SFP cooling system.

The staff's guidance for review of SFP cooling system design in SRP Section 9.1.3 specifies consideration of a single failure of a cooling system component in evaluating the capability for long-term cooling of the SFP. However, the guidance specifies that the evaluation of cooling system capability for short-term, high heat load conditions need not consider a single failure of a cooling system component. Regardless of this guidance, the licensing basis for SFP cooling is not consistent from plant to plant.

Originating Document: LER 50-245 93-011, dated 10/18/93.

Regulatory Assessment: Not all plants perform a full-core off-load during refueling. Some plants that do perform full-core off-loads have adequately addressed them in their FSARs. Plants are most at risk shortly after shutdown when decay heat is at its highest level. After some time has past (in the range of a few days) the decay heat has had a chance to dissipate to the point where SFP cooling systems can handle it. Most plants do not start moving fuel for at least a few days after shutdown. Also, although the design temperature limit for a SFP may be 150F, administrative limits in the range of 110 to 125F are in place to protect personnel working on

the refueling floor. If temperatures were to get into this administrative range during refueling, the core off-loading could simply be stopped until additional cooling could be aligned or the heat dissipated over time.

Resolution: On 11/21/95, NRC determined that development of an IN was warranted. The notice alerted licensees to the importance of assuring that: (1) planned core off-load evolutions, including refueling practices and irradiated decay heat removal, are consistent with the licensing basis, including the FSAR, TS, and license conditions; (2) changes are evaluated through the application of the provisions of 10 CFR 50.59, as appropriate; and (3) all relevant procedures associated with core offloads have been appropriately reviewed.

IN 95-54, "Decay Heat Management Practices During Refueling Outages," was issued on 12/01/95. The staff continues to look at the adequacy of SFP cooling systems and other cooling systems that are available during refueling as part of an action plan (TAC No. M88094) that began when questions were raised about the design basis of the SFP at Susquehanna Steam Electric Station.

Completion Date: 12/01/95

GCCA-0054: POTENTIAL FOR LOSS OF AUTOMATIC ENGINEERED SAFETY FEATURES ACTUATION

TAC No.: M94179 **Contact:** C. Douth

Description: On 02/02/95, during performance of an analysis of the potential effects of a steamline break such as pipe whip and jet impingement in the turbine building on the automatic ESF functions of the **W** SSPS, Pacific Gas and Electric Company (PG&E), the licensee for Diablo Canyon, identified a condition where such a break could result in failure of one train of the SSPS. This failure concurrent with a single active failure in the

other SSPS train (the plant licensing basis) could result in loss of the automatic SSPS ESF functions needed to mitigate the steamline break.

Originating Documents: 10 CFR 50.72, Event Notification 28318, dated 02/02/95, and IN 95-10, Supplement 1, "Potential for Loss of Automatic Engineered Safety Features Actuation," dated 02/10/95.

Regulatory Assessment: Subsequent to notification of the concern by PG&E, the licensees for Farley, North Anna, Salem, Sequoyah, Shearon Harris, and Summer informed the staff that their SSPS was subject to

the same concerns as those noted at Diablo Canyon. Based on the low probability of the steamline break of concern, and the fact that the manual ESF actuation capability was not affected by the postulated steamline break, the staff granted enforcement discretion for a period of two weeks to those licensees who declared the SSPS inoperable, in order for them to make the necessary modifications to fix the problem.

Resolution: In order to meet the plant licensing basis for compliance with the single failure criterion as stated in IEEE 279, the affected licensees modified the SSPS circuitry by providing qualified electrical isolation of the SSPS power supply from the input circuitry. The staff confirmed that affected licensees completed and

satisfactorily tested the modification. In addition, the staff reviewed the licensing basis for other **W** plants with the SSPS and confirmed that they are not subject to the postulated failure mode identified at Diablo Canyon.

Completion Date: 04/11/95

GCCA-0055: FAILURES IN ROSEMOUNT PRESSURE TRANSMITTERS DUE TO HYDROGEN PERMEATION INTO THE SENSOR CELL

TAC No.: M91644 **Contact:** S. V. Athavale

Description: On 11/22/94, while St. Lucie was in cold shutdown and in the process of filling and venting the RCS, two of the four pressurizer pressure channel transmitters failed high within 5 minutes of each other resulting in initiation of safety injection on high pressure signals. The affected transmitters were manufactured by Rosemount, Inc. Subsequent analysis of the failed transmitters identified entrapped hydrogen gas in the sensor cell which caused the sensor cell isolator diaphragm to deform (bow). Additional analysis determined that the isolating diaphragm material was made of Monel Alloy 400 instead of the Rosemount specified 316L stainless steel for this service. Monel 400 is permeable to monatomic hydrogen.

Originating Documents: LER 50-335, 94-009, dated 12/19/94; Rosemount 10 CFR 21, dated 03/21/95; IN

95-20, "Failures in Rosemount Pressure Transmitters Due to Hydrogen Permeation Into the Sensor Cell," dated 03/22/95.

Regulatory Assessment: The staff determined that the identified failure in Rosemount transmitters was generic, but the Rosemount Part 21 indicated only a relatively small number of affected transmitters made from a lot of Monel 400 rather than 316L stainless steel. Further, the affected transmitters are for use in high

pressure systems only. No other licensees identified similar failures in Rosemount transmitters and, because the transmitters fail high, the failure is readily detectable during calibration and would be indicated to the operator

in the control room. In addition, the failure was likely the result of gas evolution from depressurization and re-pressurization during shutdown and startup and, therefore, would not occur during normal power operation.

The staff requested all licensees to address this concern by: (1) determining whether affected Rosemount transmitters were being used in their plants; (2) assessing the impact of these transmitters on plant operational safety; and (3) identifying when the affected transmitters would be replaced with the properly designed transmitters. The staff reviewed the licensee responses and determined that proper actions were being taken to deal with the concern.

Resolution: Licensees committed to replace the affected Rosemount transmitters with correctly designed transmitters or confirmed that no affected transmitters were used in a service where they could impact plant safety. The staff conducted an inspection of the Rosemount QA process in order to confirm that a subsequent QA breakdown, such as the material error, will not occur again. The staff verified that Rosemount has taken the necessary steps to correct the QA process.

Completion Date: 04/25/95

GCCA-0056: AUGMENTED REACTOR VESSEL INSPECTION

TAC No.: M67462 **Contact:** E.J. Benner

Description: 10 CFR 50.55a (g)(6)(ii)(A), "Augmented Examination of Reactor Vessel," was issued in 1992 to require a one-time augmented inspection of the reactor vessel in accordance with the 1989 Edition of Section XI of the ASME Code. TAC No. M67462 was issued to DE to answer questions that licensees had concerning the new rule.

Originating Document: Issuance of new Rule 10 CFR 50.55a(g)(6)(ii)(A), Federal Register Notice 57 FR 34673, dated 08/06/92.

Regulatory Assessment: Based on interactions between DE and the industry on related ISI reviews, it became apparent that several licensees were either unaware of the rule or had misconceptions regarding staff approvals required by the rule when complete inspection coverage cannot be achieved. DE subsequently requested an additional TAC No. (M93643) to develop an IN to alert licensees to the presence of the rule and to provide clarification of the rule to licensees.

An IN is an appropriate generic resolution. Greater generic action is not warranted because of the pre-existing presence of the rule. The IN is warranted because of evidence that licensees are either unaware of the rule or unaware of the full ramifications of the rule demonstrated by a lack of comprehensive examinations by licensees.

Resolution: IN 96-32, "Implementation of 10 CFR 50.55a(g)(6)(ii)(A), 'Augmented Examination of Reactor Vessel,'" was issued on 06/05/96 (Accession No. 9605200277).

Completion Date: 05/29/96

GCCA-0057: CONSIDERATION OF POSITION CHANGEABLE VALVES

TAC No.: M82072 **Contact:** J.W. Shapaker

Description: The NRC, with the assistance of BNL, reviewed and evaluated the concerns associated with the mis-positioning of valves from the control room and determined that the recommendations in GL 89-10, "Safety- Related Motor-Operated Valve Testing and Surveillance," should be changed.

Originating Document: Letter from the **W** Owners Group, dated 07/21/92, requesting that the NRC notify PWR licensees that the provisions of GL 89-10 for valve mis-positioning are not applicable to PWRs.

Regulatory Assessment: The NRC no longer considers the inadvertent operation of MOVs from the control room to be within the scope of GL 89-10 for PWRs. Therefore, Supplement 7 to GL 89-10 is a relaxation of the recommendations set forth in GL 89-10 and prior supplements. Implementation of this relaxation is voluntary and Supplement 7 requests neither actions nor information from licensees. Licensees that may have taken action, or made commitments related to valve mis-positioning prior to the issuance of Supplement 7, are allowed to take advantage of this relaxed position provided licensees document the change in their GL 89-10 program.

Resolution: Issuance of GL 89-10, Supplement 7, "Consideration of Valve Mispositioning in Pressurized-Water Reactors," dated 01/24/96 (Accession No. 9601190442).

Completion Date: 01/26/96

GCCA-0058: PROBLEM OF GREASE LEAKAGE IN PRE-STRESSED CONCRETE CONTAINMENT

TAC No.: M85236 **Contact:** T.A. Greene

Description: There are 41 pre-stressed concrete containments (PCC) with greased unbonded tendons in the

U.S. The ISI requirements for PCCs provide an assurance that the grease leakage will not result in inadequate protection of tendon elements against corrosion. However, there is a concern that, if the petroleum-based grease leaks into the concrete constituents in significant amount, the concrete strength properties (compressive, shear, and bond strengths) could be reduced to an extent that the containment's capacity is appreciably degraded.

The purpose of this TAC was to investigate the properties of concrete in PCC as affected by the permeation of grease.

Originating Document: NRR became aware of significant grease leakage through the concrete of several PCCs, e.g., Trojan, ANO-1, TMI-1, and Fort Calhoun.

Regulatory Assessment: NRR is increasingly examining the effectiveness of its regulations from a risk perspective. Current regulations that govern containment design and performance are derived from assuring that the containment will withstand DBEs and provide for structural margin in the design. When evaluated under severe accident conditions, both reinforced and pre-stressed concrete containments are calculated to have ultimate failure pressure two to three times the design pressure. The results of a 1:8 scale model test of a cylindrical containment indicated a failure pressure of four times the design pressure (NUREG/CR-4209) and that of a 1:6 scale model of a cylindrical reinforced concrete containment indicated a failure pressure of three times the design pressure (NUREG/CR-5476). However, these predictions are all based on the assumption of an undegraded containment. NRC currently does not have any insights or information on the performance of a degraded containment under severe accident conditions.

Resolution: In a memorandum from W.T. Russell to D.L. Morrison, "User Needs for Degraded Containment Research," dated 05/08/96, NRR requested RES to perform research on the effects grease leakage through the containment concrete from tendon sheaths has on containment structural integrity of PCC. The objective of the proposed research is to determine the effect of leaking grease on the structure capacities of PCCs under accident conditions up to and including severe accidents. The research is to identify whether or not grease leakage leads to a loss of containment strength such that the ability of a containment with grease leakage to

meet the design basis is brought into question, or if non-negligible increase in risk occur as a result of reductions in ultimate containment capacity. If this concern is confirmed, NRR will then propose that the industry address the issue.

Completion Date: 06/17/96

GCCA-0059: GENERIC BWR STRAINER CLOGGING

TAC No.: M86925 **Contact:** J.W. Shapaker

Description: 10 CFR 50.46 requires that adequate ECCS flow be provided to maintain the core temperature at an acceptably low value and to remove decay heat for the extended period of time required by the long-lived radioactivity remaining in the core following a DBA. Therefore, based on operating experiences at several domestic and foreign reactors, the NRC issued Bulletin 96-03 to request that holders of operating licenses for

BWRs implement appropriate procedural measures and plant modifications to ensure the capability of the ECCS to perform its safety function following a LOCA. The NRC identified three potential resolution options; however, a licensee may propose an alternative approach that provides an equivalent level of assurance that the ECCS will be able to perform its safety function following a LOCA.

Originating Document: NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," published October 1995.

Regulatory Assessment: The actions requested by Bulletin 96-03 are considered backfits under the terms of 10 CFR 50.109(a)(4)(i) and are necessary to ensure that licensees are in compliance with existing NRC rules and regulations. Nevertheless, the resolution approach presented in the bulletin provides an interpretation of what licensees are expected to do to comply with 10 CFR 50.46 that heretofore has not been documented as an NRC position for the nuclear power industry.

Resolution: Issuance of Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," dated 05/06/96 (Accession No. 9605020119).

Completion Date: 05/06/96

GCCA-0060: INADEQUATE TESTING OF SAFETY-RELATED LOGIC CIRCUITS

TAC No.: M90863 **Contact:** J.W. Shapaker

Description: The NRC has issued numerous INs regarding problems with testing of safety-related logic circuits: IN 88-83, "Inadequate Testing of Relay Contacts in Safety-Related Logic Circuits," dated 10/19/88; IN 91-13, "Inadequate Testing of Emergency Diesel Generators (EDGs)," dated 03/04/91; IN 92-40, "Inadequate Testing of Emergency Bus Undervoltage Logic Circuitry," dated 05/27/92; IN 93-15, "Failure to Verify the Continuity of Shunt Trip Attachment Contacts in Manual Safety Injection and Reactor Trip Switches," dated 02/18/93; and

IN 93-38, "Inadequate Testing of Engineered Safety Features Actuation Systems," dated 05/24/93.

Despite these notices, events occurred similar to those described in the INs which indicated that licensees had not taken sufficient action to correct previously identified problems in logic circuit surveillance testing. On 03/07/95, NRC issued IN 95-15, "Inadequate Logic Testing of Safety-Related Circuits," which informed licensees about the more recent events. Nevertheless, the NRC determined that licensees should review their surveillance procedures for the RPS, EDG load-shedding and sequencing, and actuation logic for the ESFS to ensure that complete testing is being performed as required by the TS.

Originating Document: Multiple events at nuclear power reactors.

Regulatory Assessment: A number of NRC regulations document the requirements to test safety-related systems to ensure that they will function as designed when called upon. 10 CFR 50.36, Paragraph (c)(3), "Technical Specifications," states that, "surveillance requirements are requirements relating to test, calibration or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within the safety limits, and that the limiting conditions of operation will be met." Surveillance requirements to assure continued operability of safety-related logic circuits have been included in the plant-specific TS for all operating nuclear power plants. Other documents that provide a basis for these requirements include:

- 10 CFR 50.55a, "Codes and Standards," paragraph (h) which includes reference to IEEE Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations";
- Appendix A to 10 CFR 50, GDC 21, "Protection System for Reliability and Testability";
- Appendix A to 10 CFR 50, GDC 18, "Inspection and Testing of Electric Power Systems";
- Appendix B to 10 CFR 50, Criterion XI, "Test Control";
- RG 1.118, "Periodic Testing of Electric Power and Protection Systems";
- RG 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants."

Therefore, the actions requested by GL 96-01 are considered backfits under the terms of 10 CFR 50.109(a)(4)(i) and are necessary to ensure that licensees are in compliance with existing NRC rules and

regulations.

Resolution: Issuance of GL 96-01, "Testing of Safety-Related Logic Circuits," dated 01/10/96 (Accession No. 9601050193).

Completion Date: 02/27/96

GCCA-0061: BORAFLEX DEGRADATION IN SPENT FUEL POOL STORAGE RACKS

TAC No.: M91447 **Contact:** J.W. Shapaker

Description: The degradation of Boraflex that has been observed in spent fuel storage racks has been addressed by the NRC in several INs: 87-43 (Accession No. 8709010085); 93-70 (Accession No. 9309070206), and 95-38 (Accession No. 9509050009). Furthermore, EPRI has been studying the phenomenon of Boraflex degradation for several years and has identified two concerns with respect to using Boraflex in spent fuel storage racks. The first is related to gamma radiation-induced shrinkage of Boraflex and the potential to develop tears

or gaps in the material. This aspect is typically accounted for in criticality analyses of spent fuel storage racks. The second concern is the long-term Boraflex performance throughout the intended service life of the racks as a result of gamma irradiation and exposure to the wet pool environment.

Originating Document: Numerous reports of Boraflex degradation.

Regulatory Assessment: On the basis of test and surveillance information from plants that have detected areas of Boraflex degradation, no safety concern exists that warrants immediate action. Boraflex dissolution appears

to be a gradual and localized effect, forewarned by relatively high silica levels in the pool water. This occurrence of increased pool silica is more pronounced in PWRs than BWRs because of the greater effectiveness of silica removal by the BWR demineralizers in the non-borated pool water environment. Because of the safety margin present in spent fuel storage pools, compliance with the required sub-criticality margin (or conformance with the same margin to which licensees have committed in their updated FSARs) can be expected to be maintained during the initial stage of Boraflex degradation. This safety margin is due to the conservatism in treating the reactivity effects of possible variations in material characteristics and mechanical tolerances and the generally lower reactivity of stored fuel than that assumed in the safety analysis. However, to verify compliance with

both the regulatory requirements of GDC 62 and the 5% sub-criticality margins, either contained in the TS or committed to in the updated FSARs, and to maintain an appropriate degree of defense-in-depth measures, the staff has concluded that it is appropriate for licensees to submit information under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR Part 50.54(f).

Resolution: Issuance of GL 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks," dated 06/26/96 [Accession No. 9606240132].

Completion Date: 06/26/96

GCCA-0062: ANSYS AND GTSTRUDL COMPUTER PROGRAM ERROR NOTIFICATIONS

TAC No.: M91542 **Contact:** E.Y. Wang

Description: Boeing Computer Services (BCS) submitted a 10 CFR 21 notification describing errors identified in ANSYS and GTSTRUDL computer programs. Both programs are used for the design and analysis of safety-related nuclear power plant structures, systems and components, and more recently, for the design and analysis of plant modifications. There were five organizations that BCS was unable to contact because the organizations have moved or dissolved. Mail to these five organizations was returned to BCS by the US Postal Service.

Originating Document: Part 21 Log Number 95-078, a BCS Error Notification, dated 02/13/95 (Accession No. 9502220192).

Regulatory Assessment: Neither NRC nor BCS was able to evaluate the safety significance of some of the errors. Some of these errors that can prevent execution of the program sub-routines do not impact the safety-related calculations. In addition, utilities and other organizations that have received notification from BCS reported that the errors identified did not affect any safety-related calculations.

Resolution: IN 96-29, "Requirements in 10 CFR Part 21 For Reporting And Evaluating Software Errors," was issued on 05/20/96 to alert the industry of the potential problem (see Accession No. 9605150209).

Completion Date: 05/20/96

GCCA-0063: INADEQUATE CONTROL OF MOLDED-CASE CIRCUIT BREAKERS

TAC No.: M91622 **Contact:** J.R. Tappert

Description: Exercising of circuit breakers prior to surveillance testing does not provide good information on the as-found condition of the breakers. During inspections, both Diablo Canyon and South Texas licensees were found to cycle their circuit breakers prior to performing overcurrent surveillance testing. Periodic inspection and testing of circuit breakers in their as-found condition is an appropriate way of demonstrating the functional operability of the breaker and of detecting any degradation.

Originating Documents: Inspection Reports 50-275, 323/94-27, dated 12/21/94 (Accession No. 9501060003), and 50-498, 499/94-35, dated 01/19/95 (Accession No. 9502010083).

Regulatory Assessment: The practice of pre-conditioning before testing (e.g., lubricating pivot points and manually cycling the breaker) defeats the purpose of the periodic test. Such pre-conditioning does not confirm continued operability between tests, nor does it provide information on the condition of the circuit breaker for trending purposes. Testing some circuit breakers in the as-found condition can provide useful data on which to base decisions on surveillance intervals and the ability of the untested circuit breakers to perform their intended function. Since only a fraction of the circuit breakers are tested each refueling outage to justify the operability of the remaining circuit breakers, pre-conditioning before testing does not provide the expected assurance of the operability of remaining breakers which are not tested. By pre-conditioning circuit breakers, useful information may be lost because the breaker may not have been capable of performing its intended function without pre-conditioning.

Resolution: IN 96-24, "Preconditioning of Molded-Case Circuit Breakers before Surveillance Testing," was issued on 04/25/96 (Accession No. 9604220229).

Completion Date: 04/25/96

GCCA-0064: RELOCATION OF RCS PRESSURE/TEMPERATURE LIMITS

TAC No.: M91749 **Contact:** J.W. Shapaker

Description: During the development of the improved STS, a change was proposed to relocate the pressure/ temperature (P/T) limit curves and Ltop setpoint curves and values currently contained in the TS to a licensee- controlled document. As one of the improvements to the STS, the staff agreed with the industry that the curves and setpoints may be relocated outside the TS to a licensee-controlled document so that the licensee could maintain these limits efficiently and at a lower cost, provided that the parameters for constructing the curves and setpoints are derived using a methodology approved by the NRC.

Originating Document: Improved STS.

Regulatory Assessment: Any action by licensees to propose changes to TS in accordance with the guidance in GL 96-03 is voluntary and, therefore, is not a backfit under 10 CFR 50.109.

Resolution: GL 96-03, "Relocation of the Pressure/Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," was issued on 01/31/96 (Accession No. 9601290350).

Completion Date: 01/31/96

GCCA-0065: RECONSIDERATION OF PLANT SECURITY REQUIREMENTS

TAC No.: M91896 **Contact:** J.W. Shapaker

Description: In an SRM dated 02/18/94, the Commission endorsed staff recommendations to: (1) issue generic correspondence informing licensees of the opportunity to revise certain commitments in their security plan; and

(2) proceed with rulemaking regarding specific changes to reduce or eliminate certain security requirements. GL 96-02 identifies those areas in which licensees may choose to revise their plans without having to wait for the issuance of rule changes.

Originating Document: SRM dated 02/18/94.

Regulatory Assessment: The NRC has reconsidered its positions on certain security measures associated with protecting nuclear power plants against an internal threat. Suggestions contained in the GL for the reduction

or elimination of security requirements that provide only marginal protection against the insider threat are not NRC requirements, and no specific action or written response is required. Some of the suggested security plan changes require the submittal of a license amendment, in accordance with 10 CFR 50.90, while other changes may be processed in accordance with the provisions of 10 CFR 50.54(p) and can be implemented without NRC approval.

Resolution: GL 96-02, "Reconsideration of Nuclear Power Plant Security Requirements Associated With an Internal Threat," was issued on 02/13/96 (Accession No. 9601230206).

Completion Date: 02/13/96

GCCA-0066: FIRES IN EMERGENCY DIESEL GENERATOR EXCITERS

TAC No.: M92594 **Contact:** J.R. Tappert

Description: Potential for a fire in an emergency diesel exciter during operation following undetected fuse blowing. On 09/30/94 at Wolf Creek, the "A" train EDG main power potential transformer of the static exciter- voltage regulator caught fire after about one hour of sustained high power operation. On 10/11/94, the "B" EDG exciter caught fire under similar circumstances. After each fire, the licensee found that one of the 100-amp fuses in the secondary circuits of the exciter potential transformer had blown. It was subsequently determined that the fuses had blown due to manual engine shutdown without exciter shutdown. Because there was no blown-fuse indication, the normal full and above full-power runs for routine testing were conducted subsequently without knowing that the fuses had blown and "single phased" the potential transformers. The "single phased" potential transformers became overloaded, suffered progressive insulation breakdown, and eventually caught fire.

Originating Document: Inspection Report 50-482/94-13, dated 11/16/94 (Accession No. 9411220024).

Regulatory Assessment: The licensee has installed blown fuse indications on the EDG exciter cabinets. The licensee has also installed volts-per-hertz protection to avoid this potential failure mode. In other designs, under- frequency protection is often available that will independently shut down the exciter upon loss of the prime mover. However, EDG exciter systems that remain on, either through system design flaw or malfunction, after engine mechanical shutdown may fail in a manner similar to that experienced by Wolf Creek. An IN was issued to alert licensees to this concern.

Resolution: IN 96-23, "Fires in Emergency Diesel Generator Exciters During Operation Following Undetected Fuse Blowing," was issued on 04/22/96 (Accession No. 9604170169).

Completion Date: 04/22/96

GCCA-0067: INADEQUATE CAPACITY OF CCW LEADS TO FREON RELEASE TO THE CONTROL ROOM

TAC No.: M92595 **Contact:** W.F. Burton

Description: On 11/14/94, the licensee for Fort Calhoun initiated a plant shutdown because an engineering analysis had shown that the control room air-conditioners could be disabled by a large primary coolant system pipe rupture or a main steamline break inside the containment. This could result in creating an environment

in the control room which could hinder operator activities and increase temperatures above the design temperatures of safety-related equipment in the control room.

Originating Document: Event Notification 28029, dated 11/14/94.

Regulatory Assessment: A large primary coolant system pipe rupture or main steamline break inside the containment could cause the closed-cycle cooling water (CCW) temperature to rise rapidly because of the large heat input from the containment coolers during these postulated accidents. As a result, the CCW temperature could exceed the maximum post-accident CCW temperature specified in the FSAR, as well as the temperature used to calculate thermal stresses in certain piping segments.

The control room air-conditioning units are equipped with rupture discs that are designed to blow out on high CCW temperature. If the refrigerant was released, the air-conditioning units could not be recovered. Without air-conditioning, and with the control room ventilation system operating in the emergency pressurization mode,

the control room temperature could increase to levels that could hinder operator activities and cause the design temperatures of safety-related equipment in the control cabinets to be exceeded.

The complex nature of CCW systems may prevent correct identification of the most limiting potential operating configuration of the system. Certain safety-related components served by CCW systems, such as air-conditioning units and EDGs may fail in a non-recoverable manner as a result of these temperature transients. As a result of a loss of CCW, safety-related systems which depend on CCW for cooling may become inoperable. An IN has been issued to alert licensees of the need to identify the most limiting system configuration and provide the proper procurement specifications for the air-conditioning units.

Resolution: IN 96-01, "Potential for High Post-Accident Closed-Cycle Cooling Water Temperatures to Disable Equipment Important To Safety," was issued on 01/03/96 (Accession No. 9512270372).

Completion Date: 01/03/96

GCCA-0068: BWR STABILITY WITH FLOW SLIGHTLY LESS THAN NATURAL CIRCULATION FLOW

TAC No.: M92601 **Contact:** T.J. Carter

Description: Two instances of power operation have occurred in which the core flow, following run-back of the recirculation pumps, appeared to be less than that normally attributed to natural convection flow as shown on the power/flow maps available to reactor operators. Since this does not fit the expected response, the operators may not understand what could be happening to cause this observation.

Originating Document: Morning Report 1-95-0078, dated 06/06/95.

Regulatory Assessment: An apparent flow less than that associated with natural circulation may be explained by a number of subtle things. The power/flow maps are general curves and should not be assumed to present precise values, and flow instrumentation is not calibrated for flow rates near that resulting from natural convection. For these reasons, the staff does not believe there is a significant safety concern. An IN should be issued to inform the operators of the various mitigating considerations thereby clarifying their response to such an observation.

Resolution: IN 96-16, "BWR Operation With Indicated Flow Less Than Natural Circulation," was issued on 03/14/96 [Accession No. 9603110159].

Completion Date: 03/19/96

GCCA-0069: TERRY TURBINE DEPENDABILITY

TAC No.: M92636 **Contact:** T.J. Carter

Description: Over the years there have been a number of operability problems associated with Terry turbines. The turbine's principal use is as the driver for pumps in the AFW system at PWRs and in the RCIC system at BWRs, both of which are safety-related systems. Since not every licensee experiences the same problems, there has not been a universally endorsed effort to improve the operability and reliability of turbine-driven pumps. As a result, evaluations have been made of individual problems after they occur, and some facilities have implemented corrective actions. No overall assessment that considers all the known problems with universal implementation has occurred.

Originating Document: NUREG-1275, Vol. 10, "AEOD Operating Experience Feedback Report - Reliability of Safety-Related Steam Turbine-Driven Standby Pumps," October 1994, and numerous recently reported events.

Regulatory Assessment: The sequences where the turbine-driven pumps are most important to safety are those that involve a loss of all AC power, or during an SBO, particularly at PWRs. At BWRs, there are diverse or redundant means of coping with an SBO and the safety significance of a failure of a single Terry turbine is reduced. For SBO scenarios, the amount of decrease in the conditional damage frequency that can be gained by increasing the reliability of the turbine-driven AFW pump is limited by risk associated with a seal-LOCA. The perceived unreliability or unavailability of these machines is often overstated, given the fact that essentially all failures of concern are recoverable within a short period of time and, therefore, the pumps can still be available

for mitigating SBO scenarios. Thus, there is no real immediate safety concern that requires immediate regulatory response or action by the staff.

Resolution: Memorandum from L.B. Marsh to G.M. Holahan, "Terry Turbine Reliability and Future NRC Actions (TAC No. M92636)," dated 05/15/96. The conclusion was that existing and proposed rules are available to obtain needed information and achieve adequate system reliability including remedial actions. Industry should be allowed to resolve the issue through their own initiatives.

Completion Date: 06/15/96

GCCA-0070: EVALUATE IMPACT OF RCP SUPPORT COLUMN TILT ON LEAK BEFORE BREAK ANALYSES

TAC No.: M93024 **Contact:** C.V. Hodge

Description: To avert interference with crossover leg piping, the support designer at a PWR plant arranged for the base of the front inside support column for a RCP to be placed 6 to 12 inches closer to the reactor pressure vessel than in the original design. **W**, the NSSS vendor, determined that the resulting 2 to 5 degree tilt in the column decreased the vertical stiffness by a small amount. More significantly, however, **W** determined that it

might degrade the thermal expansion and loop loadings of the system. **W** identified the affected plants: Callaway, Seabrook, Watts Bar-1 & 2, Comanche Peak-1 & 2, Sequoyah-1 & 2, Wolf Creek, Farley-1 & 2, Summer,

Vogtle-1 & 2, and Shearon Harris.

Originating Document: Part 21 Log No. 95016, "Closeout of an Interim Report of an Evaluation of a Deviation or Failure to Comply Pursuant to 10 CFR 21.21(a)(2)," dated 11/15/94 (Accession No. 9411280210).

Regulatory Assessment: The safety significance of this problem is the potential decrease in analyzed safety margin against unstable crack growth in an RCS loop. The maintenance of the safety margin is needed to justify the application of leak-before-break criteria.

Resolution: EMCB reviewed the Part 21 notification for leak-before-break (LBB) analysis, determined a need for additional information from licensees of the affected plants, and requested that project managers for the affected plants request the following data: (1) comparison of applied loads used in the initial LBB analysis and the increased loads as a result of column tilt; (2) comparison of margins on critical crack size to leakage

crack size with and without the column tilt; (3) crack stability analysis using the increased piping loads; and (4) demonstration that the leakage crack will not experience unstable crack growth even if larger loads (at least $\sqrt{2}$ times the normal plus safe shutdown earthquake loads) are applied. The factor $\sqrt{2}$ may be reduced to 1.0 if the loads are combined absolutely. On the basis of information obtained, EMCB determined that all affected licensees have reviewed and responded to the **W** Nuclear Safety Advisory Letter NSAL-94-025, dated 11/10/94, (Accession No. 9607260276), on this subject. All licensees reported that the primary system piping in their plants satisfy the necessary margins in the LBB analyses.

Completion Date: 06/04/96

GCCA-0071: FISH MOUTH BURST AND BOWING OF PREVIOUSLY-PLUGGED STEAM GENERATOR TUBES

TAC No.: M93227 **Contact:** E.J. Benner

Description: During an SG inspection at Haddam Neck, a "fishmouth" opening in a plugged tube was observed. In addition, the tube was reported to be bowed towards the tube lane by 0.5 to 0.75 inches.

Originating Document: NRC Morning Report I-95-0034, dated 03/01/95.

Regulatory Assessment: **W** has previously analyzed burst plugged tube conditions because of previous events. The reported bowing of tubes is a new development in that contact of the adjacent tubes is possible. An IN has been proposed to inform licensees of **W** recommendations for burst and bowed tubes that leaking plugs be replaced, if leakage is detected during future outages, and that tubes which are adjacent to plugged tubes be inspected for signs of wear, at all future outages, until remedial action has been taken to remove the potential for plugged tubes to be pressurized to burst.

The safety significance of this concern is low because calculations of dynamic interaction of the burst tube with neighboring tubes indicate that significant wear of the neighboring tubes would not be expected in the course of one cycle of operation.

Resolution: Because of perceived low safety significance of the concern, the NRR/AEOD/RES Events Assessment Panel cancelled development of the IN with the agreement of DE on 04/02/96.

Completion Date: 03/08/96

GCCA-0072: BLOCKAGE OF UNTESTED ECCS PIPING

TAC No.: M93360 **Contact:** E.J. Benner

Description: During a refueling outage, it was discovered that two out of the four ECCS lines from the containment sump were partially blocked by debris buildup. It is presumed that the debris was present from initial construction.

Originating Document: International Reporting System Report 1505.G0 (Proprietary) dated 07/05/95.

Regulatory Assessment: Investigations by the Spanish regulatory agency revealed that several segments of ECCS piping are not subject to periodic functional flow-testing because they are not used during operation.

The potential exists for undetected blockage of these segments of pipe. This concern is potentially generic. The

safety significance of the concern is that debris in this piping can degrade system performance by reducing system flow rates and/or damaging valves, pumps, and heat exchangers.

Previous NRC generic communications (Bulletin 93-02 and Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated 05/11/93 and 02/18/94, Accession Nos. 9305110015 and 9402180174), as well as several INs have addressed aspects of this concern. Thus, an IN is the appropriate generic action to provide licensees with the additional information provided by the foreign experience.

Resolution: IN 96-10, "Potential Blockage by Debris of Safety System Piping Which Is Not Used During Normal Operation or Tested During Surveillances," was issued on 02/13/96 (Accession No. 9609070259).

Completion Date: 02/13/96

GCCA-0073: PORV INOPERABILITY MASKED BY DOWNSTREAM INDICATIONS DURING TESTING

TAC No.: M93400 **Contact:** E.J. Benner

Description: On 08/09/95, surveillance testing of the PORVs at St. Lucie indicated that they were not operating properly. The PORVs were removed and checked on a test bench. Both valves failed to relieve at any actuating delta pressure across the main valve disk to pilot vent path. The PORVs at St. Lucie are credited during feed- and-bleed and Ltop scenarios and can be used for pressure control to limit opening of SRVs.

Originating Document: Event Notification 29178, dated 08/10/95.

Regulatory Assessment: The licensee discovered that the valve guide bushings were installed backwards. The bushing has holes at one end to allow pressure beneath the main valve disk to be vented when the pilot valve

is actuated, allowing the valve to operate. With the bushing installed backwards, there was no vent path from the main valve to the pilot line. This condition probably existed since the last refueling outage (approximately 10 months ago) when a contractor, using approved licensee procedures, worked on the valves. The licensee has subsequently installed the bushings correctly, and the PORVs have tested satisfactorily.

A potential generic concern exists with the use of tailpipe acoustic monitors to verify operation of PORVs. Since the valves were refurbished, surveillance testing was performed twice with results indicating satisfactory performance of the PORVs, as evidenced by tailpipe acoustic data. The licensee subsequently determined that internal clearances in the main valve are sufficient to pass media through the pilot valve, providing positive indication on the acoustic monitors, despite the failure of the main valve to lift. There has been no previous evidence that PORVs have remained inoperable for extended periods of time because of inadequate surveillance testing. The currently proposed generic action is to issue an IN alerting licensees to the incorrectly installed bushings and the potential inadequacy of acoustic monitors to verify valve operation. An IN is the appropriate generic action to inform licensees of potential limitations of the use of acoustic monitors to verify operation.

Resolution: IN 96-02, "Inoperability of Power-Operated Relief Valves Masked by Downstream Indications During Testing," was issued on 01/05/96 (Accession No. 9512290129).

Completion Date: 01/05/96

GCCA-0074: LOSS OF RC INVENTORY AND POTENTIAL LOSS OF EMERGENCY MITIGATION FUNCTIONS WHILE IN A SHUTDOWN CONDITION

TAC No.: M93568 **Contact:** E.N. Fields

Description: On 09/17/94 at Wolf Creek, operators were attempting to re-borate RHR Train B while, at the same time, maintenance personnel were repacking an RHR Train A to Train B crossover isolation valve. Train B is re-borated by recirculating water through a loop that contains the RHR system piping, the refueling water storage tank (RWST), a containment spray pump, a manual RWST isolation valve, and a RHR system crossover line.

When the RWST isolation valve was opened for the re-boration process and the Train A to Train B crossover isolation valve was opened for stroke time testing, a drain-down path was inadvertently created from the RCS to the RWST. As a result, an unintentional RCS flow path was created allowing approximately 35,000 liters (9,200 gallons) of reactor coolant to transfer to the RWST.

Originating Document: Region IV Morning Report Number 4-94-0100.

Regulatory Assessment: If the drain-down had not been promptly terminated, the operability of the ECCS would have been compromised. Also, RCS water flashing to steam in the piping or in the RWST would likely have created conditions conducive to water hammer. This event presented an immediate safety concern.

Resolution: A Task Action Plan was developed to inform licensees and the Regions of this vulnerability, to request all licensees to take appropriate measures to prevent a similar event, and to implement a long-term resolution. As part of this Task Action Plan, IN 95-03, "Loss of Reactor Coolant Inventory and Potential Loss of Emergency Mitigation Functions While in a Shutdown Condition," was issued on 01/18/95 (Accession No. 9501110412). This IN was updated by IN 95-03, Supplement 1 (Accession No. 9602050208) of the same title. The Supplement was developed under TAC No. M93568 and describes further staff insights into this event.

Completion Date: 03/25/96

GCCA-0075: CONTROL ROD DRIVE MECHANISM PENETRATION CRACKING

TAC No.: M93641 **Contact:** E.N. Fields

Description: The activity of CRD penetration cracking has been ongoing since an incident at a foreign reactor in 1991 identified cracks in the control rod drive mechanism (CRDM) penetrations. Three U.S. licensees have voluntarily conducted inspections of CRDM penetrations, two of which have identified cracks. Approximately 20 cracks in one penetration were identified at Oconee and three cracks were identified in one penetration at

D.C. Cook. A foreign plant experienced several demineralizer resin bed intrusions which were concluded to have resulted in extensive CRDM penetration cracking. The IN will notify the industry that resin intrusions may result in accelerated corrosion of CRDM penetrations and of other components fabricated from Alloy 600.

Originating Document: Memorandum from R. Herman to A. Chaffee, dated 09/01/95.

Regulatory Assessment: There does not appear to be an immediate safety concern. No CRDM penetration failures have been experienced at U.S. reactors to date. The staff is not aware of any significant primary system resin bed intrusion at any U.S. PWR.

Resolution: IN 96-11, "Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," was issued on 02/14/96 (Accession No. 9602090038).

Completion Date: 02/14/96

GCCA-0076: AUGMENTED EXAMINATION OF REACTOR VESSEL

TAC No.: M93643 **Contact:** E.J. Benner

Description: 10 CFR 50.55a (g)(6)(ii)(A), "Augmented Examination of Reactor Vessel," was issued in 1992 to require a one-time augmented inspection of the reactor vessel in accordance with the 1989 Edition of Section XI of the ASME Code. Several licensees were unaware of the rule or had misconceptions regarding staff approvals required by the rule when complete inspection coverage cannot be achieved.

Originating Document: Issuance of new rule 10 CFR 50.55a(g)(6)(ii)(A). Federal Register Notice 57 FR 34673, dated 08/06/92.

Regulatory Assessment: TAC No. M93643 was issued to DE to develop an IN to alert licensees to the presence of the rule and to provide clarification of the rule to licensees after it became apparent that several licensees were unaware of the rule or had misconceptions regarding staff approvals required by the rule when complete inspection coverage cannot be achieved (see TAC No. M67462).

An IN is an appropriate generic resolution. Greater generic action is not warranted because of the pre-existing presence of the rule. The IN is warranted because of evidence that licensees are either unaware of the rule or unaware of the full ramifications of the rule.

Resolution: IN 96-32, "Implementation of 10 CFR 50.55a(g)(6)(ii)(A), `Augmented Examination of Reactor Vessel,'" was issued on 06/05/96 (Accession No. 9605200277).

Completion Date: 06/05/96

GCCA-0077: CLOSED HEAD VENT CAUSES INACCURATE LEVEL INDICATION DURING REDUCED INVENTORY

TAC No.: M93751 **Contact:** R.A. Benedict

Description: Closed head vent causes inaccurate level indication during reduced inventory. On 09/13/95, Surry-1 was in a shutdown condition, cooled down and depressurized preparatory to refueling. A pressurizer PORV and its block valve had been opened, connecting the pressurizer to the pressurizer relief tank (PRT) which was pressurized with nitrogen to 11 psig. The reactor head vent was open to the top of the water-level indicating standpipe through the vent connection to the PORV relief line. The water level in the reactor and pressurizer had been lowered to slightly below the reactor vessel flange so that the vessel head studs could be de-tensioned. The pressurizer was empty and the reactor coolant piping was full to part way up the surge line.

In order to install the cavity seal ring so that the cavity could be flooded up to permit lifting the reactor head, the head vent valve was closed and the ventline spoolpiece was disconnected. After the seal ring was in place, the spoolpiece was re-connected but the head vent valve was not reopened as it should have been. This resulted in loss of function of the only reactor water level indication available while the reactor head was still installed.

Letdown and makeup were being maintained manually by an operator who monitored the standpipe level. As pressurizer relief tank overpressure was being reduced, the operator saw an increase in indicated level due to water in the reactor vessel being forced up into the standpipe and surge line as the gas bubble trapped in the reactor vessel expanded. The operator increased letdown to maintain the indicated level required. This process continued periodically for about five hours, resulting in reactor coolant inventory being reduced by almost 5,000 gallons.

When the relief tank pressure was subsequently reduced to atmospheric pressure and the reactor vessel studs were de-tensioned, the vessel head lifted enough to relieve the gas pressure in the head. This caused a sudden drop in indicated level in the standpipe by about five feet. The operator immediately took action to restore the level to where it was supposed to be.

Originating Document: NRC Region II Morning Report Number 2-95-0083, dated 09/01/95.

Regulatory Assessment: The safety significance of this particular event is minimal: the water level remained

1.5 feet above mid-loop and more than six feet above the core. Forced circulation residual heat removal continued. If the gas bubble had continued to expand, the water level would not have dropped lower than just below the top of the reactor coolant hot-leg pipe; at that point, the gas bubble would vent through the hot-leg pipe into the pressurizer and into the pressurizer relief tank, equalizing pressure between the relief tank and the vessel atmosphere. The standpipe indicated level would then have been correct.

Previous events have occurred in which gas bubbles and level indication problems have been of concern. These events have been the subjects of previously-issued INs. An IN is the appropriate generic action to inform licensees of considerations involved in this present Surry event.

Resolution: IN 96-37, "Inaccurate Reactor Water Level Indication and Inadvertent Draindown During Shutdown," was issued on 06/18/96 (Accession No. 9606120154).

Completion Date: 06/18/96

GCCA-0078: SHUTDOWN COOLING FLOW BYPASSING CORE RESULTS IN TEMPERATURE AND PRESSURE INCREASES

TAC No.: M93752 **Contact:** C.V. Hodge

Description: On 10/03/95, the Events Assessment Panel classified the set of two undetected mode changes of the BWR at Hope Creek as a Significant Event for the NRC Performance Indicator Program. The basis for this classification is the failure of the Hope Creek licensee to comprehend the condition of the plant for an extended period of time.

Originating Document: Preliminary Notification of Event or Unusual Occurrence, PNO-I-95-026, "Improper Reactor Recirculation System Alignment Led to Shutdown Cooling Flow Bypassing the Reactor Core," dated 08/02/95.

Regulatory Assessment: On 08/09/95, the licensee reported that a shutdown cooling bypass event had occurred at 1100 hours on 07/08/95, when the operating crew left the "B" recirculation loop discharge valve in a partially open position to mitigate potential thermal binding of that valve. During the shutdown cooling evolution, 2,000 gpm of RHR flow was diverted through the open recirculation valve and redirected from the intended path (through the core) to a parallel path (through recirculation Loop "B"). This parallel path bypassed the core. In addition, operators later secured the RHR system, in accordance with plant procedures, to facilitate testing

of the shutdown cooling isolation valves. The 2,000 gpm bypass flow combined with the isolation of the RHR system resulted in the heatup of the RCS that caused the first undetected mode change. Shutdown cooling was then returned to service. About ten hours later, bypass flow increased to approximately 4,000 gpm when the recirculation valve was further opened in an attempt to re-close it, causing the second undetected mode change. The valve was manually closed at 0550 hours on 07/09/95, terminating the event. Licensee investigation into this event identified key corrective actions in the areas of operator training, operator procedure compliance, valve thermal binding assessment, and management response to the event. The licensee determined on 08/04/95

that an operational condition change occurred from cold shutdown to hot shutdown (the undetected first mode change); this was not known at the time of the event. As a result of both these unplanned mode changes, several TS LCOs were not met.

The NRC conducted a special inspection of circumstances surrounding this event and concluded that this event was initiated when plant operators inappropriately left open the recirculation pump discharge valves, allowing shutdown cooling flow to bypass the reactor vessel, which decreased the ability of the shutdown cooling system to remove decay heat and allowed the RCS temperature and pressure to increase. This resulted in

an undetected change in the plant operational condition from the desired cold shutdown to the hot shutdown condition (the undetected first mode change).

Resolution: The inspection team concluded that the principal causes of this event were inadequate communications and failure to follow procedures and that contributing causes were poor quality procedure instructions and inadequate training. In addition, senior plant management initially failed to correctly assess the significance of this event. The failure resulted in a 10-day delay in initiating a comprehensive root cause

evaluation and contributed to the failure to make the required notification to the NRC. The team also concluded that this event was safety significant. However, the consequences of the event were minimal and the event had no direct adverse effect on the health and safety of the public or plant personnel. The identified weakness in both operator and management performance during and following this event were also significant. On 09/13/95, the Events Analysis and Generic Communications Branch (PECBB) briefed senior NRC management on the event and, on 10/05/95, PECBB and Region I briefed the ACRS. On 01/18/96, IN 96-05, "Partial Bypass of Shutdown Cooling Flow from the Reactor Vessel," was issued describing this event.

Completion Date: 01/19/96

GCCA-0079: POTENTIAL CONTAINMENT LEAK PATH THROUGH HYDROGEN ANALYZER

TAC No.: M93753 **Contact:** J.R. Tappert

Description: Hydrogen analyzers communicate with the containment atmosphere after an accident and can create a containment leak path. Two plants recently identified potential containment leak paths through the hydrogen monitor system. At Braidwood, a potential containment bypass existed for several months when a hydrogen sensing line was not re-connected following an integrated containment leak rate test. At Catawba, the hydrogen analyzer was tested in a de-energized condition. After an accident, the analyzer would be energized and the pressure boundary changes. The testing would not reveal any leakage in the energized pressure boundary.

Originating Document: LER 50-457/95-02-01, dated 04/21/95 (Accession No. 9504240323).

Regulatory Assessment: Because containment penetrations, systems, and equipment that will be exposed to the containment atmosphere must be leak rate-tested to ensure that containment integrity is

maintained after

a DBA, the procedures for these tests must adequately consider the penetration configuration. Additionally, because hydrogen monitor containment isolation valves are normally procedurally opened after a DBA, any leakage in the hydrogen monitor system may bypass the containment and can challenge regulatory radiological guidelines. An IN was issued to alert licensees to these concerns.

Resolution: IN 96-13, "Potential Containment Leak Paths Through Hydrogen Analyzers," was issued on 02/26/96 (Accession No. 9602220234).

Completion Date: 02/26/96

GCCA-0080: INADEQUATE TESTING AND DESIGN OF TORNADO DAMPERS

TAC No.: M93754 **Contact:** T. Koshy

Description: On 03/02/94, South Texas-1 reported that a rapid depressurization could occur in the HVAC ducts in the event of a tornado. A design deficiency limited the closing of a damper. On 10/22/93, River Bend reported in LER 93-020 that a design deficiency could result in loss of ventilation to several buildings after passage of a tornado.

Originating Documents: River Bend Station LER 93-020, dated 10/22/93; South Texas Unit 1 LER 94-003, dated 03/02/94.

Regulatory Assessment: Even though Tornado dampers were not tested, only one of the 30 dampers failed to operate when tested at South Texas Project. The problem at River Bend was the failure of dampers to open

when the fans are running. This problem appears to be generic, since an accident condition is not considered to exist while these dampers are challenged, a reasonable time is available to manually manipulate the dampers and recover the affected systems. Therefore, an IN is adequate response to build the awareness in the industry and for overcoming this vulnerability.

Resolution: IN 96-06, "Design and Testing Deficiencies of Tornado Dampers at Nuclear Power Plants," was issued on 01/25/96 (Accession No. 9601190306).

Completion Date: 01/26/96

GCCA-0081: ASSESSMENT OF CORROSION OF B&W FUEL USED IN 2-YEAR FUEL CYCLES

TAC No.: M93842 **Contact:** E.J. Benner

Description: TMI-1 shut down on 09/08/95 for its scheduled refueling and maintenance outage. The licensee found an unusual build-up of corrosion products on approximately 40 of the 177 fuel assemblies. Some of the fuel assemblies with the corrosion product buildup have clad wall thinning of the outer face pins. The licensee inspections revealed small defects in a total of nine fuel pins in five assemblies. The pin-hole size defects in the pins' metal cladding have occurred only in first cycle fuel with high (4.75%) enrichment.

Originating Document: NRC Morning Report I-95-0126, dated 09/27/95.

Regulatory Assessment: The licensee believes that the corrosion and associated defects are due to particulars of this fuel cycle, which was the first two-year cycle of the unit. The particulars include highly enriched fuel (to allow a two-year cycle with the lower enrichment once-burned and twice-burned fuel), and the high RCS boron concentration and high power peaking factors associated with highly enriched fuel. Mid-cycle, the licensee instituted a lithium addition program to combat corrosion found in the CRD area (also attributed to the high boron concentration) and believes that this will reduce future corrosion of the fuel assemblies.

The safety significance of this type of fuel cracking is very likely bounded by the conditions that existed during this cycle, since the corrosion was limited to "peaking" fuel assemblies and future cycles will have lower peaking factors. This concern may be generic as licensees migrate to two-year fuel cycles. The NRC will continue to follow the licensee corrective action and take action as necessary. TAC No. M93842 was opened for SRXB to evaluate the need for additional generic action.

Resolution: SRXB determined that additional follow-up activity was necessary. RVIB has been tasked with inspecting the B&W fuel fabrication facility to evaluate manufacturing processes and use of Codes to predict fuel failures during extended cycles.

Completion Date: 02/02/96

GCCA-0082: ENVIRONMENTAL EFFECTS ON MAIN STEAM SAFETY VALVE SET POINT

TAC No.: M94004 **Contact:** J.R. Tappert

Description: On 09/22/95, ANO-2 began testing their main steam safety valves (MSSV) using a Crosby lift assist device. The first valve tested did not lift at a simulated pressure of 6.12% over the nominal set pressure. The licensee stopped testing, reviewed their procedures, and the following day resumed testing. The next valve tested lifted at 4.27% over nominal and the third valve would not lift at 5.9% over nominal. At this point, the licensee stopped testing and developed a detailed action plan. ANO subsequently removed all 10 MSSVs and shipped them to Wyle Laboratories for testing and/or refurbishment.

On 09/30/95, Wyle tested one of the valves ANO was unable to lift in-situ. The valve lifted at 0.97% above nominal which was within the acceptance range. The licensee had personnel onsite at Wyle and immediately

began to investigate the discrepancy between the two test results. The difference appears to be caused by the differences in the environments in which the valves were tested. The in-situ test was done with uninsulated valves with an ambient temperature of approximately 95F. The Wyle test was done with insulated valves

in an ambient environment of 140F. When the valve was retested at Wyle attempting to replicate the ANO environment, the safety valve lifted at 5.65% above nominal, much closer to the in-situ observed conditions. Wyle then went on to test all of the MSSVs using simulated ANO environmental conditions. Five of the ten valves failed to meet the TS required tolerances of +1%/-3%, with two valves exceeding +6%.

Originating Document: LER 50-368/95-05, dated 11/02/95 (Accession No. 9511080301).

Regulatory Assessment: Because the licensee did not provide adequate guidance to its vendor, the MSSVs were mis-calibrated. The licensee subsequently determined that using actual as found values for the safety valves, they could show protection for all DBEs. It was not clear how many licensees might also have a discrepancy between their testing and operational environments, so an IN was issued to inform them of ANO's experience.

Resolution: IN 96-03, "Main Steam Safety Valve Setpoint Variation as a Result of Thermal Effects," was issued on 01/05/96 (Accession No. 9512290299).

Completion Date: 01/15/96

GCCA-0083: INADVERTENT DRAINING OF REACTOR VESSEL AND ISOLATION OF SHUTDOWN COOLING SYSTEM

TAC No.: M94044 **Contact:** N.K. Hunemuller

Description: In November 1995, Hatch-2 was in its twelfth refueling outage in the cold shutdown mode; the "A" loop of the RHR system was in the shutdown cooling mode. In accordance with the recent licensee implementation of improved STS, component operation from the remote shutdown panel was being tested for the first time. When maintenance and operations personnel performed activities to determine the cause of deficiencies identified during the testing, approximately 12,000 gallons of water drained out of the reactor vessel in less than 1 minute. The low level of water in the reactor vessel triggered automatic isolation of the

shutdown cooling system, terminating the event. Further investigation revealed that an interlock designed to prevent a drain-down had been set improperly, actually causing the event. Although the event was compounded by personnel, procedural, and maintenance errors, NRC inspectors attributed the root cause to inadequate modification, maintenance, and testing control with respect to the remote shutdown panel and related equipment.

Recent implementation of improved STS at various utilities may result in surveillance tests using circuitry that previously went unchallenged. Over time, these circuits may have degraded or were modified and caused unexpected performance. The normal plant configuration may not be the most desirable configuration for these new tests. For example, the normal control switch line-up on the remote shutdown panel may be an appropriate line-up for mitigating a control room fire but may be less appropriate for testing individual components. The licensee operational experience described in this IN highlights the importance of plant configuration control when implementing new surveillance tests.

Originating Document: Event Notification 29548, dated 11/02/95.

Regulatory Assessment: The safety significance of this particular event appears to be limited. The shutdown cooling containment isolation functioned as designed and multiple ECCS makeup sources were operable as required by TS. Reactor vessel water level remained above the top of active fuel. However,

an IN highlighting both the speed of the draindown and plant configuration control for tests involving operations from the remote shutdown panel was determined to be warranted.

Resolution: IN 96-15, "Unexpected Plant Performance During Performance of New Surveillance Tests," was issued on 03/08/96 (Accession No. 9603040234).

Completion Date: 03/13/96

GCCA-0084: RECENT PROBLEMS WITH OVERHEAD CRANES

TAC No.: M94045 **Contact:** J.R. Tappert

Description: Problems with overhead cranes had been identified at two different sites. At Trojan, a section of the reactor building polar crane bridge rail failed due to the inappropriate flame-cutting of bolt slots during initial construction. At Prairie Island, the overhead crane handling system inappropriately automatically stopped on overload while lifting a loaded spent fuel storage cask. The crane stopped due to an inaccurately calibrated overload-sensing system.

Originating Documents: Inspection Reports 50-344/95-06, dated 09/18/95 (Accession No. 9509210219), and 50-282/95-06, dated 06/27/95 (Accession No. 9507070029).

Regulatory Assessment: Crane failures adversely affect plant operations and could lead to a radiological accident. An IN was promulgated to inform licensees of these recent problems.

Resolution: IN 96-26, "Recent Problems with Overhead Cranes," was issued on 04/30/96 (Accession No. 9604260095).

Completion Date: 04/30/96

GCCA-0085: REMOVING REFUELING FLOOR SHIELDING PLUGS PRIOR TO AND SOON AFTER SHUTDOWN

TAC No.: M94088 **Contact:** E.Y. Wang

Description: Oyster Creek was planning to move the shield plugs before shutdown. Weighing 10 to 85 tons, the shield plugs are layered directly above the reactor vessel. The primary purpose of shield plugs is to provide protection to plant personnel working on the refueling floor. Oyster Creek wanted to save outage time by removing the shield plugs before the reactor was shutdown. Other utilities have similar practices.

Originating Document: Region I SMM Pre-brief in October 1995.

Regulatory Assessment: A concern was raised regarding moving shield plugs before a plant is shut down, just before a refueling outage begins. There are several sites that have such practice, including Oyster Creek, Limerick, and Millstone-1; Cooper has recently started the same practice. Safety concerns have been raised regarding personnel safety under accident conditions and under LOCA conditions, after the shield plugs are removed. There is also a concern of dropping the heavy load of shield plugs, a condition which has not been analyzed at some of the plants. At one site, this activity was precluded by the FSAR; however, this activity was done under 10 CFR 50.59. A question exists whether 10 CFR 50.59 adequately considered all of the above safety concerns.

Resolution: The resolution of this concern was included in the Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel Pool, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," issued on 04/11/96 (Accession No. 9604080259). TAC M94912 was issued to capture the effort on this bulletin. The subject of the

bulletin was to address concerns of moving heavy loads, including dry casks and other heavy loads over reactor vessel and spent fuel pool. Since the concern with shield plugs was addressed in this bulletin, no separate generic communication was deemed necessary.

Completion Date: 05/07/96

GCCA-0086: DAMAGE TO VALVE INTERNALS CAUSED BY THERMALLY-INDUCED PRESSURE LOCKING

TAC No.: M94189 **Contact:** T.J. Carter

Description: Observed damage to an internal component of a valve was attributed to thermally-induced

pressure locking. A retaining ring had been bent; this was indicative of an internal pressure between 3000 and 7000 psi. Valves adjacent to piping systems subjected to large temperature changes could be susceptible to thermally-induced pressure locking, inoperability, and possible damage.

Originating Document: Event Notification 29659, dated 11/30/95.

Regulatory Assessment: Temperature increase of fluid trapped in valve bonnets can cause very high pressures. Potential pressure locking and thermal binding is being addressed in GL 95-07. A number of improbable factors, such as a heat source sufficient to actually raise the trapped fluid temperature and a valve that is leak-tight enough to contain the resulting pressure increase, are necessary to achieve very high

pressures. An IN should be prepared that alerts licensees to the potential for an undetected thermally-induced pressure increase in valves.

Resolution: IN 96-08, "Thermally Induced Pressure Locking of a High Pressure Coolant Injection Gate Valve," was issued on 02/05/96 (Accession No. 9601300092).

Completion Date: 02/05/96

GCCA-0087: DAMAGE IN FOREIGN STEAM GENERATOR INTERNALS

TAC No.: M94254 **Contact:** E.J. Benner

Description: In April 1995, during a routine eddy current inspection of the SG tubing at a foreign facility, anomalous support plate signals were observed at the uppermost support plate. The SGs are similar to Model 51 SGs. The support plates are of the drilled-hole type and are fabricated from carbon steel. Video camera inspections were conducted to investigate the anomalous signals and revealed that a significant portion of the support plate had wasted away. Pieces of the affected region of the support plate were found resting on the next lower support plate. Subsequent investigation identified chemical cleaning performed in 1992 as the cause of the support plate damage.

Originating Document: Bilateral agreement with a foreign country.

Regulatory Assessment: The SG tube support plates function to support the tubes against lateral displacement and vibration and to minimize bending moments in the tubes during accidents. Known instances of support plate cracking/damage in the U.S. have generally involved support plates with significant denting. The potential for support plate cracks has tended to not be of significant concern in recent years since the SGs most affected

by denting have been replaced and the industry has been successful in controlling denting progression. The foreign experience serves to highlight that there are other mechanisms which may lead to support plate cracking/ damage.

Based on the information available to the staff, it is not yet known whether SGs in the U.S. are vulnerable to the type of wrapper damage observed at the foreign unit. The staff will continue to monitor information on support plate and wrapper damage as it becomes available from foreign authorities. Issuance of an IN is commensurate with the known safety significance and applicability of the concern. Factors impacting priority determination were the potentially high safety significance and the lack of direct available evidence indicating applicability to U.S. plants.

Resolution: IN 96-09, "Damage in Foreign Steam Generator Internals," was issued on 02/12/96 (Accession No. 9602060170).

Completion Date: 02/12/96

GCCA-0088: INTERFACE BETWEEN OPERATORS AND NUCLEAR ENGINEERS DURING TESTS AND STARTUP

TAC No.: M94370 **Contact:** E.M. McKenna

Description: During individual control rod testing in March 1995, Dresden-3 briefly exceeded the TS limit for maximum fuel design limiting ratio for centerline melt (FDLRC). The nuclear engineer recognized that the limit might be exceeded when a particular (high-worth) rod was withdrawn, but also knew that subsequent insertion of the rod would restore the ratio. The potential exceedance was not communicated to the operators. The licensee investigation showed weaknesses in its reactivity control program such as incorrect focus on pre-conditioning limits and failure to perform predictor model calculations when conditions changed. Further, it was noted that operations personnel heavily relied on the nuclear engineers and lacked sufficient knowledge to question thermal limit trends.

Originating Document: Dresden Unit 3 (50-249) LER 95-05-01, dated 11/16/95 (Accession No. 9511210148).

Regulatory Assessment: The potential exists that facility personnel other than NRC-licensed operators may effectively control reactivity manipulations under certain circumstances. An example is a nuclear engineer supervising rod testing, if the licensed operator is positioning rods without having the knowledge or procedures necessary to ensure compliance with applicable TS limits. The safety significance of the specific exceedances at Dresden was low due to the small amount of exceedance and the short duration. However, the Events Assessment Panel authorized issuance of a TAC No. for long-term follow-up of potential generic concerns about the interface between operations and nuclear engineering for reactivity control situations.

Resolution: The Operator Licensing Branch had conducted a review of the adequacy of licensee control of reactivity changes during startup and during rod pattern manipulations as part of another task. As part of this review, an informal survey of reactor engineering and operations practices was performed, and a search of LER data bases. Based on this review, risk insights, and the indication/ protection systems available, it was concluded that generic action by the NRC is not necessary (see memo from S. Richards to A. Chaffee on 02/05/96).

Completion Date: 02/05/96

GCCA-0089: VALVE STEM COUPLING OF GIMPEL AUXILIARY FEEDWATER TURBINE TRIP THROTTLE VALVES

TAC No.: M94371 **Contact:** T.J. Carter

Description: Reports were received that identified 2 mechanisms for disengagement of linkages used in the operation of turbine governor controls. One involved missing or improperly installed set screws that would prevent unscrewing of a coupling. The other mechanism involved the use of too "thick" a lock nut whose locking mechanism was not engaged.

Originating Documents: Morning Reports 3-94-0146 and 4-94-0102, Event Notification 29111, and a vendor letter dated 08/30/94 (Accession No. 9409150014).

Regulatory Assessment: When the connecting coupling between the valve stem and operator becomes loose, erratic operation of the control valve is observed. This could impact operation of the turbine throttle valve used in the AFW system of PWRs, a safety-related system. The defect also could impact BWRs, both the RCIC and HPCI systems. The priority for resolution was judged to be moderate even though the probability of failure is believed low based upon the number of observed failures. Based on follow-up information, the importance was less than originally perceived. By this time, the coupling deficiency had been addressed by the vendor notifying their customers.

Resolution: Issuance of an IN was cancelled. The coupling concern was addressed by the vendor.

Completion Date: 04/30/96

GCCA-0090: IMPROPER EQUIPMENT SETTINGS DUE TO THE USE OF NON-TEMPERATURE COMPENSATED TEST EQUIPMENT

TAC No.: M94468 **Contact:** E.N. Fields

Description: The use of non-temperature compensated test gauges to calibrate and test safety-related equipment was identified at Farley and Surry. Non- temperature compensated gauges were used in environments that required that temperature corrections be applied to gauge readings. However, licensees were not correcting gauge readings.

Originating Document: Memorandum from E. Mershoff to D. Crutchfield, dated 11/20/95.

Regulatory Assessment: No immediate safety concern was identified, i.e., no instances were identified where TS setpoints were exceeded; however, a potential existed that a limit could be exceeded. Systems potentially affected included reactor trip system transmitters, MSSV lift settings, ESFAS transmitters, and pressure instruments used for calorimetric calculations. Therefore, an IN was determined to adequately address this issue.

Resolution: IN 96-22, "Improper Equipment Settings Due to the Use of Nontemperature-Compensated Test Equipment," was issued on 04/11/96 (Accession No. 9604050336).

Completion Date: 05/07/96

GCCA-0091: USE OF INDIVIDUAL PLANT EXAMINATIONS (IPEs) FOR REGULATORY DECISION MAKING

TAC No.: M94469 **Contact:** N.K. Hunemuller

Description: An IN was proposed to restate the objectives of IPEs, including the IPEEEs, to address the purpose of the staff review of the IPE and IPEEE submittals, and to address the potential relationship of the IPE and IPEEE program to other ongoing and future regulatory programs. The proposed IN was to inform licensees that the use of information from an IPE submittal for purposes other than those associated with GL 88-20 would likely require additional staff review.

Originating Document: Memorandum from A. Thadani, "Use of Individual Plant Examinations (IPEs) for Regulatory Decision Making," dated 10/03/95 (Accession No. 9510100018).

Regulatory Assessment: The majority of licensees have indicated their intention to update and maintain their IPEs (i.e., PRAs) and to use these PRAs in regulatory applications beyond GL 88-20. This use of PRA is encouraged in the Commission's Policy Statement on "Use of Probabilistic Risk Assessment Methods in

Nuclear Regulatory Activities" and the staff is pursuing activities directed toward greater use of risk information in regulatory decision-making. However, the staff review of the IPEs has not been performed for this purpose. Therefore, to help ensure that the scope (and limitations) of the staff's review of the IPE submittals is clearly understood, guidance has been provided to the staff to clarify that the staff review was not designed to provide the sole basis for risk-informed regulatory decision-making and that use of a licensee's IPE for regulatory decisions other than associated with GL 88-20 will likely require additional staff review. A letter to the NEI was issued on 04/24/96 to provide similar guidance to licensees.

Resolution: On 04/19/96, a Commission Paper entitled "Clarification of IPE/IPEEE Objectives, Staff Review Purpose, and Potential Future Regulatory Uses of IPE/IPEEE" was issued with a letter to the NEI attached. On 04/24/96, the letter to W. Rasin (NEI) from A. Thadani (NRR) was issued.

Completion Date: 03/04/96

GCCA-0092: OVERWITHDRAWAL OF TIP

TAC No.: M94470 **Contact:** E.Y. Wang

Description: On 10/31/95, LaSalle-1 experienced difficulty with overwithdrawal of the TIP outside its shield and shield room. This resulted in high radiation levels in portions of the reactor building. The licensee declared an alert based on the radiation presenting a potential over-exposure of plant staff.

Originating Document: Event Notification 29259, dated 10/31/95.

Regulatory Assessment: The TIP system has total of five channels; "1B" was withdrawn beyond the shielded storage location in the reactor building. The LaSalle TS requires at least four operable drive machines and that TIP data for an inoperable measurement location may be replaced by data obtained from a 3-dimensional BWR core simulator code normalized with available operating measurements. Since the other four TIPs were operable, there is no operational problem in with this event. There is an area radiation monitor in the vicinity of TIP drive machines which provide an alarm in the control room. In addition, LaSalle personnel uses electronic dosimeters which also provide alarm when radiation level reaches certain setpoint.

Resolution: IN 96-25, "Traversing In-Core Probe Overwithdrawn at LaSalle County Station, Unit 1," was issued on 04/30/96 (Accession No. 9604250172).

Completion Date: 04/30/96

GCCA-0093: SPENT FUEL POOL COOLING

TAC No.: M94480 **Contact:** D.L. Skeen

Description: The adequacy of SFP cooling at nuclear power plants was called into question when it was discovered that Millstone-1 routinely performed full-core off-loads during refueling, even though the plant's licensing basis described a partial-core off-load during normal refueling. As part of the subsequent Task Action Plan on Spent Fuel Storage Pools (TAC No. M88094), all NRR project managers were directed to perform a survey of the current licensing basis for the SFP at each plant. TAC No. M94480 was issued to track the time project managers spent conducting the survey. The results of the survey were submitted to SPLB for evaluation.

Originating Document: Memorandum from the EDO to the Chairman concerning lessons learned from Millstone Unit 1, dated 12/28/95 (Accession No. 9603120370).

Regulatory Assessment: After evaluating the survey results, the staff determined that the existing structures, systems, and components related to storage of irradiated fuel provide adequate protection for public health and safety. However, the staff review identified strengths and weaknesses and potential areas for safety enhancements for individual plants.

Resolution: The concern was resolved when all project managers submitted their survey results to SPLB (see TAC No. M88094 for resolution of SFP issues).

Completion Date: 05/31/96

GCCA-0094: SOUTH TEXAS STUCK ROD EVENT FOLLOWING REACTOR TRIP

TAC No.: M94494 **Contact:** S.S. Koenick

Description: On 12/18/95, with South Texas-1 at 100% power, a pilot wire monitoring relay actuation caused a main transformer lock-out which resulted in a turbine trip and reactor trip. In response to the reactor trip, three control rod bottom lights failed to light and the digital rod position indicated six steps out for each rod. Following the transient, one rod drifted to the bottom and the other two were manually inserted. During subsequent rod testing, the three control rods and an additional control rod failed to fully insert.

South Texas has a 14-foot core with **W** Standard XL, Standard XLR, and VANTAGE 5H 17 x 17 fuel assemblies, and the affected control rods were found in twice-burned Standard XLR fuel with burnup greater than 42,880 megawatt days(MWD)/metric ton uranium(MTU).

Originating Document: Event Notification 29734, dated 12/18/95.

Regulatory Assessment: On a plant-specific basis, the transient was within the plant design basis and all systems effectively functioned as designed. With respect to the rods stopping six steps from the bottom, the rod worth for the last six steps is minimal and the licensee verified adequate shutdown margin.

On a generic application of control rod problems, the item is significant in that stuck control rods could affect safe shut down margin following design basis transients. For South Texas, the licensee performed a safety evaluation that supports negligible impact to shutdown margin and reload safety evaluation assuming 32 control assemblies do not insert below 12 steps following a reactor trip. Therefore, there appears to be justification for continued operation while the root cause is being pursued.

Resolution: IN 96-12, "Control Rod Insertion Problems," dated 02/15/96 (Accession No. 9602090161), discussed details of both the South Texas trip and the Wolf Creek trip on 01/30/96. Subsequently, Bulletin 96-01 (Accession No. 9603120001) was issued on 03/08/96 requesting **W** utilities to conduct specified control rod tests. The root cause of the control rod problem is still under investigation.

Completion Date: 02/15/96

GCCA-0095: RADWASTE FACILITY EQUIPMENT DEGRADATION AT MILLSTONE UNIT 1

TAC No.: M94521 **Contact:** E.Y. Wang

Description: During an NRC routine inspection, it was identified that a portion of the radwaste processing facility was apparently not maintained; the waste storage tanks had indications of leaks, there was indication of a few feet of flooding in the waste storage room; corrosion of piping and tanks in the facility was evident; and waste build-up and equipment damage were also observed.

Originating Document: Inspection Report 50-245/95-35, dated 09/11/95 (Accession No. 9509180214).

Regulatory Assessment: The radwaste room condition has degraded significantly because of the waste build-up and equipment damage. The radwaste room is designed to have limited access. The radwaste storage tank has degraded to a condition that it no longer could store the waste. The whole room became a storage place for the radwaste. The operational safety significance concern is minimal. There is no specific NRC regulation regarding the radwaste room conditions. Yet, NRC issued IE Circular No. 80-18 on 08/22/80 (Accession

No. 8006190038), "10 CFR 50.59 Evaluations For Changes To Radioactive Waste Treatment Systems,"

which requires licensee to perform a safety evaluation in accordance with 10 CFR 50.59 for the following circumstances: (1) components described in the SAR are removed; (2) component functions are altered; (3) substitute components are utilized; or (4) changes remain following completion of maintenance activity.

Resolution: IN 96-14, "Degradation of Radwaste Facility Equipment at Millstone Nuclear Power Station, Unit 1," was issued on 03/01/96 (Accession No. 9602260117).

Completion Date: 03/04/96

GCCA-0096: WOLF CREEK REACTOR TRIP WITH ONE TRAIN ESSENTIAL SERVICE WATER SYSTEM INOPERABLE

TAC No.: M94594 **Contact:** J.R. Tappert

Description: On 01/30/96, operators at Wolf Creek received alarms indicating that the circulating water system traveling screens were becoming blocked. A visual inspection showed that the traveling screens for Bays 1 and 3 were frozen and that water levels in these bays were approximately 8 ft below normal. The essential service water system was started with the intent to separate this system from the service water system. At approximately 3:30 a.m., operators received a service water pressure alarm and an electric fire pump started on low service water pressure. The shift supervisor then directed a manual reactor/turbine trip. Circulating water system bays were subsequently determined to be at 12 feet below normal. The level loss was caused by water from the spray wash system freezing and blocking the traveling screens.

The Train A essential service water system pump was tripped and declared inoperable at 7:47 a.m. because of low discharge pressure and high strainer differential pressure. At about 5:45 p.m., the operators declared Train A operable on the basis of an engineering evaluation and placed it in service. However, the pump was

again stopped 1.5 hours later at approximately 7:30 p.m. when the pump exhibited further oscillations in flow and pressure. At approximately 8:00 p.m., operators noted that essential service water system Train B suction bay level was 15 ft below normal and decreasing slowly. Operators placed additional heat loads on Train B and the suction bay levels subsequently recovered.

At about 10:00 a.m. on 01/31/96, divers inspected the suction bay of Train A and noted complete blockage of the trash racks by frazil ice. Train B was not inspected because the pump was running. The ice blockage was cleared by 4:00 p.m. by sparging the trash racks with air. The essential service water system was designed to have warming flow injected in front of trash racks to increase bulk water temperature and prevent the formation of frazil ice. Due to calculational errors by the architect-engineer and the as-built system configuration, the essential service water system warming flow was insufficient to prevent frazil ice from forming at the Train A trash racks.

Originating Documents: Event Notification 29904 and 29905 dated 01/30/96, LER 50- 482/96-01 (Accession No. 9603120274), and LER 50-482/96-02 (Accession No. 960310619).

Regulatory Assessment: Facility vulnerability to icing events is a function of plant design. Frazil and other ice formation is dependent on specific environmental conditions and represent a potential common-mode failure that can cause the loss or degradation of multiple cooling water systems, including the potential loss of the UHS. Loss of the UHS is potentially significant and it was not clear what facilities would be vulnerable to this failure mode. Therefore, an IN was issued to alert licensees to recent ice-related events.

Resolution: IN 96-36, "Degradation of Cooling Water Systems Due to Icing," was issued on 06/12/96 (Accession No. 9606070097).

Completion Date: 06/12/96

GCCA-0097: STUCK CONTROL ROD PROBLEMS

TAC No.: M94608 **Contact:** J.R. Tappert

Description: PWR control rods fail to completely insert upon a scram signal. On 12/18/95, with South Texas-1 at 100% percent power, an electrical transient led to a reactor trip. While verifying that control rods had inserted fully after the trip, operators noted that the rod bottom lights of 3 control rod assemblies were not lit; the digital rod position indication for each rod indicated 6 steps withdrawn. A step is equivalent to 1.59 cm [5/8 inch] and the top of the dashpot begins at 38 steps. During subsequent testing of all control rods in the affected banks, the rod position indication for the same 3 locations as well as a new location indicated 6 steps withdrawn. As compared to prior rod drop testing, no significant differences in rod drop times were noted before reaching the upper dashpot area for any of the control rods. All 4 control rods were located in fuel assemblies that were in their third cycle with burnup greater than 42,880

megawatt days per metric ton uranium (MWD/MTU).

On 01/30/96, after a manual scram from 80% power, 5 control rod assemblies at Wolf Creek failed to insert fully. Two rods remained at 6 steps withdrawn, 2 at 12 steps, and 1 at 18 steps. At Wolf Creek, a step is equivalent to

1.59 cm [5/8 inch] and the top of the dashpot begins at approximately 30 steps. Three of the affected rods drifted to fully inserted within 20 minutes, 1 within 60 minutes, and the last one within 78 minutes. The results also indicate that there was some slowing down of affected rods before reaching the dashpot. During subsequent cold drop tests, the same 5 rods plus an additional 3 rods failed to fully insert. All of the affected rods were in 17x17 VANTAGE 5H fuel assemblies with burnup greater than 47,600 MWD/MTU.

On 02/21/96, during the insert shuffle in preparation for loading North Anna-1 Cycle 12, 2 new control rods assemblies could not be removed with normal operation of the handling tool from the fuel assemblies in the spent fuel pool in which they were temporarily stored. The control rod assemblies were removed using the rod assembly handling tool in conjunction with the bridge crane hoist. The 2 affected fuel assemblies were VANTAGE 5H assemblies which had achieved 47,782 MWD/MTU and 49,613 MWD/MTU burnup during 2 cycles of irradiation.

Originating Documents: PNO-IV-95-059 and Event Notification 29904, dated 01/30/96.

Regulatory Assessment: The events discussed earlier, as well as several similar events at foreign reactors, raise concerns about the operability of control rods in high burnup fuel assemblies. While most of the testing to date has demonstrated that the control rods have reached the dashpot region of the guide tube and that adequate shutdown margin has been maintained, there have been indications of degraded rod drop times and

a stuck rod well above the dashpot region. Thus, there is concern that these events may be precursors to more significant control rod binding problems in which required shutdown margins and rod drop times may be violated. Due to the fact that the control rod binding mechanism was not fully understood a Bulletin was issued to alert licensees to these events and request that licensees with **W**-designed plants assess control rod operability and verify the control rod drop times, rod recoil, and drag forces at the next scheduled shutdown (EOC, maintenance, etc.) for all rodded fuel assemblies.

Resolution: Bulletin 96-01, "Control Rod Insertion Problems," was issued on 03/08/96 (Accession No. 9603120001). Also, IN 96-12, "Control Rod Insertion Problems," was issued on 02/15/96 (Accession No. 9602090161).

Completion Date: 03/13/96

GCCA-0098: FAILURE OF TONE ALERT RADIO TO ACTIVATE WHEN RECEIVING A SHORTENED ACTIVATION SIGNAL

TAC No.: M94768 **Contact:** J.B. Birmingham

Description: During an Emergency Response Test at Callaway, the length of the tone alert signal time was found to be insufficient to activate some portions of the Tone Alert Network.

Originating Document: IN Authorization Request Form presented to the Events Assessment Panel on 02/13/96.

Regulatory Assessment: This concern has a moderate degree of safety significance and generic applicability in that tone alert radio signals are typically part of an overlapping system designed to alert the general populace in the event of needed emergency action. Although the FCC authorized the time length of the signal to be reduced, many stations have not made any changes. Additionally, the failure of the tone to activate tone alert radios could be detected during emergency response tests. However, the potential failure of the signal to activate tone alert radios reduces the overall performance of the alert response system.

Resolution: IN 96-19, "Failure of Tone Alert Radio to Activate when Receiving a Shortened Activation Signal," was issued on 04/02/96 (Accession No. 9603270127).

Completion Date: 04/02/96

GCCA-0099: SLOW FIVE PERCENT SCRAM INSERTION TIMES CAUSED BY VITON DIAPHRAGMS IN SCRAM SOLENOID PILOT VALVES

TAC No.: M94778 **Contact:** D.L. Skeen

Description: Degradation of the control rod 5% scram insertion times has been noted at some BWR plants. GE and the manufacturer, Automatic Switch Company (ASCO), have determined that the cause of the slow times is adherence of the fluoroelastomer (Viton) diaphragm to the brass valve seat of the scram solenoid pilot valve (SSPV).

Originating Document: Event Notification 29879 from Brunswick Unit 1, dated 01/23/96.

Regulatory Assessment: Scram time limits are imposed by TS to ensure that the control rods will be inserted into the reactor core fast enough to prevent exceeding the minimum critical power ratio (MCPR) and, thus, prevent damage to the fuel. At Brunswick, the TS limit for average core-wide insertion to notch 46 (or 5% into

the core) for all control rods is 0.358 seconds. Scram data taken by the licensee following the manual scram on 01/23/96 showed that the TS limits were exceeded (actual core-wide average was about 0.4 seconds).

The event scenario of concern is a reactor trip at end of core life without turbine bypass valves opening. At the end of core life all control rods are fully withdrawn from the core (Position 48) and the flux pattern is shifted to the top of the core. Thus, a delay in scram time could potentially cause some fuel to be damaged. However, GE performed a safety analysis and determined that exceeding the 5% insertion time would not result in core damage as long as the other TS-required scram insertion times (20%, 50%, and 90%) were met. None of these other insertion times have been exceeded as a result of the diaphragm sticking to the seat.

Resolution: IN 96-07, "Slow 5% Scram Insertion Times Caused by Viton Diaphragms in Scram Solenoid Pilot Valves" was issued on 01/26/96 (Accession No. 9601260139). After meeting with the NRC on 01/26/96, the BWR Owners Group activated their Regulatory Response Group to resolve the issue. GE and the BWROG worked with ASCO to develop an improved formulation of Buna-N rubber as an interim measure to replace the Viton diaphragm. The Buna-N diaphragm became available in June 1996. GE is currently developing a

composite diaphragm that will include the original Viton as the outer portion because of its superior elastic quality with a center portion made of a much harder Viton to prevent the diaphragm from sticking to the valve seat. GE hopes to have the composite diaphragm available by late August 1996.

Completion Date: 01/26/96

GCCA-0100: POTENTIAL CLOGGING OF HPSI THROTTLE VALVES DURING CONTAINMENT SUMP RECIRCULATION PHASE

TAC No.: M94808 **Contact:** E.J. Benner

Description: Northeast Utilities determined that eight manual throttle valves in the high-pressure safety injection lines were susceptible to clogging during the recirculation phase of a LOCA at Millstone-2. The licensee based this determination on the fact that the openings in the containment sump screens are 0.187" and the minimum dimension within the valve flow path is 0.125".

Originating Document: Event Notification 29999, dated 02/20/96.

Regulatory Assessment: The manual throttle valves are inaccessible during an accident. The safety significance is exacerbated by the fact that the normal lineup for recirculation at this unit has the low-pressure safety injection pump feeding the HPSI system, with all recirculation flow passing through the HPSI system. The licensee has adopted this arrangement because of structural and vibrational concerns with the LPSI system.

This concern is generic. The Millstone concern was discovered because of licensee review of a similar concern at Diablo Canyon. The concern at Diablo Canyon was dispositioned as not safety significant because the screen was sufficient to prevent a clogging problem. The appropriate NRC action is to expedite issuance of an information notice, in addition to continued evaluation of the issue. No additional generic action is necessary because of several mitigating factors including: (1) particles passing through the sump strainers may be pulverized by the high-pressure safety injection pumps; (2) differential pressure across the valve may force debris through the valve; (3) other plants may have sump strainer openings smaller than the valve opening; (4) debris may settle in the sump at the flow rates involved with high pressure recirculation; and (5) post-accident recirculation lineup (i.e., whether all recirculation flow must pass through the HPSI system).

Resolution: IN 96-27, "Potential Clogging of High Pressure Safety Injection Throttle Valves During Recirculation," was issued on 05/01/96 (Accession No. 9604260077).

Completion Date: 05/01/96

GCCA-0101: STEAM GENERATOR TUBE INSPECTION RESULTS

TAC No. M94862 **Contact:** E.J. Benner

Description: SG tube examinations have been performed at a number of plants during the last year. As a result of these examinations, degradation has been observed at a number of locations including dented locations, the expansion transition region, free span region, and in the tubesheet crevice.

In addition to identifying a variety of degradation mechanisms, a number of technical concerns have arisen as a result of these examinations with respect to classifying inspection results, periodicity of examinations, and expansion of the initial inspection scope.

Originating Documents: Several Event Notifications including: 28446 (Braidwood), dated 03/02/95; 28482 (Maine Yankee), dated 03/04/95; 28646 (Kewaunee), dated 04/07/95; 29494 (Diablo Canyon), dated 10/22/95; and 29571 (Byron), dated 11/07/95.

Regulatory Assessment: In general, the degradation modes observed have been consistent with past experience; however, the results indicate the importance of performing comprehensive SG tube examinations using appropriate inspection techniques. SGTRs can provide a direct release path for contaminated primary coolant to the environment via the secondary side safety and relief valves. Accumulation of water in the

SG secondary side can also lead to an overflow condition which can severely aggravate the radiological consequences and increase the likelihood of subsequent failures. An IN is an appropriate generic action to inform licensees of the particulars of recent examination methodologies and failure mechanisms.

Resolution: IN 96-38, "Results of Steam Generator Tube Examinations," was issued on 06/21/96 (Accession No. 9606180338).

Completion Date: 06/21/96

GCCA-0102: REACTOR OPERATION BELIEVED TO BE INCONSISTENT WITH THAT DESCRIBED IN THE FSAR

TAC No.: M94911 **Contact:** T.J. Carter

Description: Licensees may not be maintaining and operating their facilities in compliance with their licenses and their bases. A staff follow-up to a 10 CFR 2.206 Petition involving Millstone-1 discussed this situation. A self-assessment was performed by Northeast Utilities Service Companies (Accession No. 9603150021) in response to a staff order issued on 12/13/95 (Accession No. 9512150278).

Originating Documents: A petition from Galatis and Hadley, dated 08/21/95 (Accession No. 9509080209), and its Supplement, dated 08/28/95 (Accession No. 9509110306).

Regulatory Assessment: The staff has a concern about Northeast Utilities' performance regarding operation, controlling facility changes, and maintaining an accurate updated FSAR for Millstone-1. The licensee's self-assessment stated that a potential exists that similar configuration management conditions may exist at several of their other units. To alert other licensees to what may be a generic deficiency, an IN should be issued that transmits both the Millstone licensee's self evaluation and the staff 10 CFR 50.54(f) letter that expresses the concern.

Resolution: IN 96-17, "Reactor Operation Inconsistent With The Updated Final Safety Analysis Report," was issued on 03/18/96 (Accession No. 9603150213).

Completion Date: 03/18/96

GCCA-0103: MOVEMENT OF DRY STORAGE CASKS OVER SPENT FUEL, FUEL IN THE REACTOR CORE, OR SAFETY-RELATED EQUIPMENT

TAC No.: M94912 **Contact:** E.Y. Wang

Description: A licensee was planning to move an unanalyzed load along a path where a load drop would have significant safety consequences.

Originating Document: See TAC No. M94088.

Regulatory Assessment: The licensee's 10 CFR 50.59 evaluation was based on a misunderstanding of the purpose of GL 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants'

NUREG-0612." The staff determined that this was an unreviewed safety question because: (1) the casks were heavier than those previously considered in the FSAR; and (2) a load drop could result in consequences that are greater than previously evaluated in the FSAR and, therefore, the margin of safety could be reduced.

Resolution: Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel Pool, Over Fuel in the Reactor Core, or Over Safety-related Equipment," was issued on 04/11/96 (Accession No. 9604080259). The bulletin was to address concerns of moving heavy loads, including dry casks, over the reactor vessel and the spent fuel pool. Since the concern with shield plugs (TAC No. M94088) was addressed in this bulletin, no separate generic communication was deemed necessary.

Completion Date: 05/07/96

GCCA-0104: INACCURACY OF DIAGNOSTIC EQUIPMENT FOR MOTOR-OPERATED BUTTERFLY VALVES

TAC No.: M95281 **Contact:** T.A. Greene

Description: ITI MOVATS Inc. developed the Butterfly Analysis and Review Test (BART) System as a method for determining the torque output of Limatorque HBC gear boxes equipped with Limatorque motor actuators

on butterfly valves. Observations and questions concerning the performance of the BART System under field conditions led ITI MOVATS to perform testing to determine more precisely the inaccuracy of the system. This TAC No. was for the issuance of an IN to alert licensees to the increased inaccuracy of the BART diagnostic equipment for measuring torque when operating butterfly valves.

Originating Document: Morning Report 4-96-0042, dated 04/24/96.

Regulatory Assessment: Testing has raised questions concerning the inaccuracy assumed for ITI MOVATS BART diagnostic equipment for butterfly valves. The inaccuracy could be as high as 14% compared to the previous 2% error assumption. The increased inaccuracy could adversely affect safety-related butterfly valves set up with the diagnostic equipment.

Resolution: IN 96-30, "Inaccuracy of Diagnostic Equipment for Motor-Operated Butterfly Valves," was issued on 05/21/96 (Accession No. 9605160311).

Completion Date: 05/21/96

GCCA-0105: CROSS-TIED SAFETY INJECTION ACCUMULATORS

TAC No.: M95282 **Contact:** J.R. Tappert

Description: Many licensees may have operated with cross-tied SI accumulators in an unanalyzed condition. On 03/08/96, the licensee for Indian Point-3 (IP-3) reported that they had operated outside of their design basis because they had periodically cross-tied SI accumulators for short periods of time. IP-3 TS actually require periodic cross-connecting under certain conditions. An evaluation by the licensee's engineering staff (confirmed by **W**) shows that the plant may not be protected during some LOCAs with a cross-tied configuration. This is because nitrogen pressure is postulated to bleed off through the faulted loop to the containment. After the IP-3 report, several other licensees including IP-2, Turkey Point-3 & 4, Byron-1 & 2, Braidwood-1 & 2, Zion-1 & 2, and Vogtle-1 & 2 reported that their plant procedures allow cross-connection of SI accumulators (in some cases, all of the accumulators) in order to equalize pressure. No other licensee reported a requirement to perform this operation.

Originating Documents: Event Notification 30364, dated 04/25/96, initiated action, but the first notification was from IP-3 Event Notification 30087, dated 03/08/96.

Regulatory Assessment: IP-3 performed a probabilistic evaluation to determine if the configuration exceeded the EPRI screening criteria of 10E-6 core damage probability. The licensee concluded that, for the bounding time estimate of 11 hours/year for cross-connecting two SI accumulators, the EPRI screening threshold was not met. IP-3 has taken actions to change their TS and other licensees have taken administrative action to prohibit cross- connecting the accumulators. Therefore, this concern appears to be generic, but the safety significance is limited and the licensees who have been made aware of the problem have taken corrective action.

Resolution: IN 96-31, "Cross-Tied Injection Accumulators," was issued on 05/22/96 (Accession No. 9605170288) to alert licensees to the potential problem.

Completion Date: 05/22/96

GCCA-0106: HYDROGEN GAS IGNITION DURING WELDING OF A VSC-24 MULTI-ASSEMBLY SEALED BASKET

TAC No.: M95483 **Contact:** T.A. Greene

Description: A hydrogen generation and ignition event occurred at Point Beach during the welding of the shield lid on a spent fuel storage cask. The hydrogen was generated by a chemical reaction between the cask materials and the borated spent fuel water. A review of previous cask loadings at Point Beach and other plants indicates that this situation has occurred before with casks of the same design. This TAC was initiated for the issuance of an IN to alert licensees to the Point Beach event.

Originating Document: Event Notification 30552.

Regulatory Assessment: The Point Beach event raised concerns about the potential for chemical or other reactions between cask materials, contents, and environments in spent fuel storage cask designs. The basic concern is that these reactions may create hazardous operating conditions and degrade cask safety components to the extent that the cask's ability to store fuel safely will be compromised.

Resolution: IN 96-34, "Hydrogen Gas Ignition During Closure Welding of a VSC-24 Multi-Assembly Sealed Basket," was issued on 05/31/96 (Accession No. 9605310132).

Completion Date: 05/31/96³

³ SECY-81-513, "Plan for Early Resolution of Safety Issues," U.S. Nuclear Regulatory Commission, August 25, 1981. [8109140067]

