Appendix D: Related Generic Activities

This appendix documents those activities related to generic issues, i.e., related generic activities (RGA) that did not meet the criteria for designation as generic issues (GI) but were important enough to require the development of Action Plans by NRR to address the concerns. The plan for documenting these RGAs was

delineated in SECY-96-107.[1] On this page:

GA-001: BOILING WATER REACTOR INTERNALS DESCRIPTION

Significant cracking of the core shroud was first observed at Brunswick-1 in September 1993. The NRC notified licensees of Brunswick's discovery of significant circumferential cracking of the core shroud welds. In 1994, core shroud cracking continued to be the most significant of reported internals cracking. In July 1994, the NRC issued

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

GL 94-03[2] which required licensees to inspect their shrouds and provide an analysis justifying continued operation until inspections could be completed.

A special industry review group (Boiling Water Reactor Vessels and Internals Project - BWRVIP) was formed to focus on resolution of reactor vessel and internals degradation. This group was instrumental in facilitating

licensee responses to GL 94-03.[3] The NRC evaluated the BWRVIP reports, submitted in 1994 and early 1995, and all plant responses. All of the plants evaluated were able to demonstrate continued safe operation until inspection or repair on the basis of: (1) no 360° through-wall cracking observed to date; (2) low frequency of pipe breaks; and (3) short period of operation (2 to 6 months) before all of the highly susceptible plants complete repairs of, or inspections to, their core shrouds.

In late 1994, extensive cracking was discovered in the top guide and core plate rings of a foreign reactor which was similar in design to GE reactors in the U.S.; however, there have been no observations of such cracking

in U.S. plants. GE concluded that it was reasonable to expect that the ring cracking could occur in GE BWRs that had been in operation for more than 13 years. In the BWRVIP report that was issued in January 1995, ring cracking was evaluated. The NRC concluded that the BWRVIP assessment was acceptable and that top

guide ring and core plate ring cracking was not a short-term safety issue. This activity was identified in an NRR memorandum[4] to RES in February 1996.

Many components inside BWR vessels, i.e., internals, are made of materials such as stainless steel and various alloys that are susceptible to corrosion and cracking. This degradation can be accelerated by stresses from temperature and pressure changes, chemical interactions, irradiation, and other corrosive environments.

The action plan[5] developed by NRR is intended to encompass the evaluation and resolution of issues associated with IGSCC in BWR internals, including plant-specific reviews and the assessment of the generic criteria proposed by the BWR Owners' Group. The staff will continue to assess the scopes that have yet to be submitted by licensees concerning inspections or re-inspections of their core shrouds. The staff will also continue to assess core shroud inspection results and any appropriate core shroud repair designs on a case- by-case basis. The staff will issue separate safety evaluations regarding the acceptability of core shroud

inspection results and core shroud repair designs. The staff has been interacting with the BWRVIP and individual licensees. In an effort to lower the number of industry and staff resources that will be needed in the future, it is important for the staff to continue interacting with the industry on a generic basis in order to encourage them to continue their proactive efforts to resolve IGSCC of BWR internals. The BWRVIP has submitted four generic documents, supporting plant-specific submittals, for staff review. The staff is ensuring that the generic reviews are incorporating recent operating experience on all BWR internals.

CONCLUSION

Based on licensee responses to GL 94-03,[6] the staff concluded in all cases that licensees have provided

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[2] Generic Letter 94-03, "Intergranular Stress-Corrosion Cracking of Core Shrouds in Boiling Water Reactors," U.S. Nuclear Regulatory Commission, July 25, 1994. [ML031070204]
[3] Generic Letter 94-03, "Intergranular Stress-Corrosion Cracking of Core Shrouds in Boiling Water Reactors," U.S. Nuclear Regulatory Commission, July 25, 1994. [ML031070204]
[4] Memorandum for C. Serpan from A. Chaffee, "Nuclear Reactor Regulation (NRR) Input Into Research NUREG-0933 (WITS Item 9400213)," February 13, 1996. [9602260124]
[5] Memorandum for A. Thadani from B. Sheron, "Staff Action Plan for the Resolution of Issues Associated with Boiling Water Reactor Internals Cracking," April 26, 1995. [9505220070]
[6] Generic Letter 94-03, "Intergranular Stress-Corrosion Cracking of Core Shrouds in Boiling Water Reactors," U.S. Nuclear Regulatory Commission, July 25, 1994. [ML031070204]

sufficient evidence to support continued operation of their BWR units to the refueling outages in which shroud inspections or repairs have been scheduled. In addition, by the end of 1995, industry's special review group that is aggressively pursuing this issue was expected to issue a comprehensive plan addressing cracking in all

BWR internals, discussing cracking susceptibility, safety consequences, inspection scope and methodology, flaw evaluation, repair strategies, and mitigation of degradation.

Almost all BWRs will have completed inspections or repairs of core shrouds during refueling outages by the fall of 1995. Various repair methods have been used to provide alternate load-carrying capability, including preemptive repairs, installation of a series of clamps and use of a series of tie-rod assemblies. The NRC has reviewed and approved all shroud modification proposals that have been submitted by BWR licensees. Review by NRC continues on individual inspection results and plant-specific assessments.

The industry special review group was expected to issue a comprehensive plan addressing cracking in all BWR internals, discussing cracking susceptibility, safety consequences, inspection scope and methodology, flaw evaluation, repair strategies, and mitigation of degradation. The NRC is reviewing new information submitted by GE on the safety significance of, and recommended inspections for, top guide and core plate ring cracking.

GA-002: REACTOR PRESSURE VESSEL FRACTURE TOUGHNESS DESCRIPTION

As a result of the information provided by licensees in response to GL 92-01,[7] Revision 1, issued in March

1992, the staff issued NUREG-1511[8] and the Reactor Vessel Integrity Database (RVID). NUREG-1511[9] provides a summary of the critical issues and regulatory requirements involved in RPV structural integrity and the status of each RPV with respect to the regulatory requirements. The RVID contains all the data that were submitted by licensees to demonstrate compliance with the regulatory requirements. Since licensees provide

data during the life of their plants to demonstrate compliance with regulatory requirements, NUREG-1511[10] and the RVID will require periodic upgrading.

In April 1995, the staff completed its evaluation of the Palisades plant compliance with the PTS Rule (10 CFR 50.61) and concluded that the Palisades RPV could be operated in compliance with the requirements of the PTS Rule through the plant's 14th refueling outage scheduled for late 1999. To extend the life of the Palisades

RPV beyond 1999, the licensee has begun to plan for annealing of the RPV. The Palisades reactor vessel will be

the first commercial nuclear vessel annealed in the U.S. to improve its fracture toughness. The staff will review the licensee's annealing plan prior to its implementation scheduled for the 1998 refueling outage. Prior to this anneal, the industry will perform demonstration anneals at the Marble Hill and Midland-2 sites. This activity was identified in an NRR memorandum[11] to RES in February 1996.

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[7] Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants (Except Yankee Atomic Electric Company, Licensee for the Yankee Nuclear Power Station) from U.S. Nuclear Regulatory Commission, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f) (Generic Letter 92-01)," February 28, 1992, (Rev. 1) March 6, 1992 [ML031070438], (Rev. 1, Supplement 1) May 19, 1995 [ML031070449].

[8] NUREG-1511, "Reactor Pressure Vessel Status Report," U.S. Nuclear Regulatory Commission, December 1994.

[9] NUREG-1511, "Reactor Pressure Vessel Status Report," U.S. Nuclear Regulatory Commission, December 1994.

[10] NUREG-1511, "Reactor Pressure Vessel Status Report," U.S. Nuclear Regulatory Commission, December 1994.

[11] Memorandum for C. Serpan from A. Chaffee, "Nuclear Reactor Regulation (NRR) Input Into Research NUREG-0933 (WITS Item 9400213)," February 13, 1996. [9602260124]

10 CFR 50, Appendix G, and 10 CFR 50.61 establish requirements to prevent fracture of the RPV and require licensees to project the amount of embrittlement of RPV materials. As a result of the review of responses to GL

92-01,[12] the review of the Palisades PTS, and inspections conducted at CE offices in Windsor, Connecticut, several concerns related to RPV evaluations have been identified and are summarized as follows:

(1) It appears that licensees may not have been aware of, or considered, all relevant information and data in previous assessments of their RPVs;

(2) The variability in copper and nickel chemical composition may be independent of weld heat number and is greater than previously recognized by the staff.

Based on the above findings, the staff concluded that the most effective way to resolve the concern was through a supplement to GL 92-01[13] requiring licensees to collect all data relevant to their RPVs and, if there are

data that had not been previously considered, to perform a reassessment of their RPVs. As a result of the data supplied in response to GL 92-01[14] and the Palisades PTS review, NRR requested RES to evaluate whether changes to the PTS Rule or RG 1.99[15] are necessary.

Specific actions included in the generic action plan[16] are: (1) issue Supplement 1 to GL 92-01[17]; (2) coordination with RES on RPV integrity issues; (3) hold an NRC/Industry workshop on RPV issues; (4) review first and second round of responses to GL 92-01,[18] Supplement 1; (5) issue Supplements 1 and 2 to NUREG-1511[19]; (6) issue Revisions 1 and 2 of the RVID; (7) observe industry annealing

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[12] Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants (Except Yankee Atomic Electric Company, Licensee for the Yankee Nuclear Power Station) from U.S. Nuclear Regulatory Commission, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f) (Generic Letter 92-01)," February 28, 1992, (Rev. 1) March 6, 1992 [ML031070438], (Rev. 1, Supplement 1) May 19, 1995 [ML031070449].

[13] Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants (Except Yankee Atomic Electric Company, Licensee for the Yankee Nuclear Power Station) from U.S. Nuclear Regulatory Commission, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f) (Generic Letter 92-01)," February 28, 1992, (Rev. 1) March 6, 1992 [ML031070438], (Rev. 1, Supplement 1) May 19, 1995 [ML031070449].

[14] Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants (Except Yankee Atomic Electric Company, Licensee for the Yankee Nuclear Power Station) from U.S. Nuclear Regulatory Commission, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f) (Generic Letter 92-01)," February 28, 1992, (Rev. 1) March 6, 1992 [ML031070438], (Rev. 1, Supplement 1) May 19, 1995 [ML031070449].

[15] Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, July 1975, (Rev. 1) April 1977 [7907100362], (Rev. 2) May 1988. [8907270187]

[16] Memorandum for A. Thadani from J. Strosnider, "Plan for Addressing Generic Reactor Pressure Vessel Issues," August 9, 1995. [9508150078]

[17] Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants (Except Yankee Atomic Electric Company, Licensee for the Yankee Nuclear Power Station) from U.S. Nuclear Regulatory Commission, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f) (Generic Letter 92-01)," February 28, 1992, (Rev. 1) March 6, 1992 [ML031070438], (Rev. 1, Supplement 1) May 19, 1995 [ML031070449].

[18] Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants (Except Yankee Atomic Electric Company, Licensee for the Yankee Nuclear Power Station) from U.S. Nuclear Regulatory Commission, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f) (Generic Letter 92-01)," February 28, 1992, (Rev. 1) March 6, 1992 [ML031070438], (Rev. 1, Supplement 1) May 19, 1995 [ML031070449].

[19] NUREG-1511, "Reactor Pressure Vessel Status Report," U.S. Nuclear Regulatory Commission, December 1994.

demonstrations; (8) review and evaluate the Palisades annealing plan; and (9) review the Palisades anneal.

CONCLUSION

The staff's assessment[20] of the impact of increased variability in chemistry on the RT_{PTS} value of PWR vessels indicated that there is no immediate cause for concern and that there is adequate time to perform a more rigorous assessment of the issue. GL 92-01,[21] Supplement 1 was issued and an NRC/Industry workshop[22] was held. Requests[23][24] [25] for research on RPV integrity were made by NRR and the RVID was issued (NRC Administrative Letter 95-03[26]).

GA-003: DRY CASK STORAGE OF SPENT FUEL DESCRIPTION

Since 1986, several U.S. nuclear power plant licensees have installed independent spent fuel storage installations (ISFSIs), i.e., licensee-owned dry cask storage facilities; other licensees are also planning such installations. In recent years, licensees have encountered a number of problems during the fabrication, installation, and licensing of some of these ISFSIs and there has been an inconsistent level of performance by involved licensees and cask fabricators with respect to the use of dry cask storage of spent reactor fuel. Because of the anticipated increased industry effort in this area, the staff needed to fully understand the problems that occurred and take appropriate measures to reduce such problems in the future. Therefore, NMSS and NRR reviewed the lessons learned by all Offices and the Regions from past experience with ISFSIs and developed a plan[27] to resolve major concerns and problems; this plan was to be pursued in accordance with

an MOU[28] between NRR and NMSS. This activity was identified in an NRR memorandum[29] to RES in February 1996.

The concern addresses dry storage of fuel that is several years old. Technical concerns have been addressed on a site-specific basis for existing facilities. The action plan will improve guidance, enhance communications with industry and the public, and aid future applicants.

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[20] Memorandum for A. Thadani from J. Strosnider, "Assessment of Impact of Increased Variability in Chemistry on the RTPTS Value of PWR Reactor Vessels," May 5, 1995 [9505100187]

[21] Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants (Except Yankee Atomic Electric Company, Licensee for the Yankee Nuclear Power Station) from U.S. Nuclear Regulatory Commission, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f) (Generic Letter 92-01)," February 28, 1992, (Rev. 1) March 6, 1992 [ML031070438], (Rev. 1, Supplement 1) May 19, 1995 [ML031070449].

[22] Memorandum for L. Shao from M. Mayfield, "Summary, NRC/NEI Workshop on Nuclear RPV Integrity," September 6, 1995. [9509200141]

[23] Memorandum for E. Beckjord from W. Russell, "NRR User Need Request for Support of Resolving Problem of Stress Corrosion Cracking of Reactor Vessel Internal Components," December 2, 1994. [9505090299]

[24] Memorandum for E. Beckjord from W. Russell, "NRR User Need Request for Support of Resolving Problem of Stress Corrosion Cracking of Reactor Vessel Internal Components," December 2, 1994. [9505090299]

[25] Memorandum for D. Morrison from W. Russell, "Request for Research on Reactor Pressure Vessel Integrity," August 11, 1995. [9508220323]

[26] Administrative Letter 95-03, "Availability of Reactor Vessel Integrity Database," U.S. Nuclear Regulatory Commission, August 4, 1995. [ML031110318]

[27] Memorandum for J. Taylor from C. Paperiello and W. Russell, "Dry Cask Storage Action Plan," July 28, 1995. [9508250186]

[28] Memorandum for J. Taylor from W. Russell and R. Bernero, "Realignment of Reactor Decommissioning Program," March 15, 1995. [9508250180]

[29] Memorandum for C. Serpan from A. Chaffee, "Nuclear Reactor Regulation (NRR) Input Into Research NUREG-0933 (WITS Item 9400213)," February 13, 1996. [9602260124]

The Action Plan[30] was developed to identify and resolve major concerns and problems in the area of dry cask storage of spent reactor fuel in ISFSIs. Specific concerns encompassed by the plan include heavy load control, procedures for cask loading and unloading, failed fuel storage, change processes, inspection activities, and communications (internal and external). Concerns have been divided into the following categories: near-term technical; long-term technical; communications; and process issues. Actions included in the plan are: (1) review each general issue and identify the specific problems to be addressed; (2) develop corrective actions for each problem; and (3) implement the corrective actions.

CONCLUSION

The following action plan items have been completed: cask trunnions; hydrostatic testing; cask weeping; and 10 CFR 72 reporting requirements. The Regions, NMSS, and NRR hold regular interface calls to discuss dry cask issues, training has been given to many of the affected staff, and NRC has established open communications with the newly-formed Nuclear Energy Institute Dry Cask Storage Working Group.

GA-004: THERMO-LAG FIRE BARRIERS DESCRIPTION

In June 1991, NRR established a special team to review the safety significance and generic applicability of technical issues regarding the use of Thermo-Lag fire barriers. In April 1992, the special review team issued its final report[31] which identified concerns about fire endurance, combustibility, and ampacity derating.

Subsequently, NRR prepared an Action Plan[32] to address the issues associated with Thermo-Lag and the NRC fire protection program. This plan included evaluation and resolution of generic Thermo-Lag fire barrier concerns regarding toxicity, construction and installation, fire endurance, ampacity derating, combustibility, seismic capabilities, and uniformity of materials. The plan also called for the staff to evaluate the special review team findings and public concerns, coordinate with NEI and licensees, conduct fire endurance and ampacity derating tests, and assess the NRC reactor fire protection program. This activity was identified in an NRR memorandum[33] to RES in February 1996.

In response to Bulletin 92-01[34] and its Supplement, licensees with Thermo-Lag fire barriers established NRC- approved measures, such as fire watches, to compensate for possible inoperable fire barriers. The combination of compensatory measures and the defense-in-depth fire protection features provide an adequate level of fire protection until licensees implement permanent corrective actions.

Specific actions in the Action Plan[35] include: (1) the resolution of generic concerns raised by the special review team; and (2) resolution of plant-specific issues that emerge from the generic concerns.

CONCLUSION

In June 1994, the Commission approved a staff recommendation to resolve Thermo-Lag concerns by

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[30] Memorandum for J. Taylor from C. Paperiello and W. Russell, "Dry Cask Storage Action Plan," July 28, 1995. [9508250186]

[31] Memorandum for W. Russell from T. Murley, "Final Report"Special Review Team for the Review of Thermo-Lag Fire Barrier Performance," April 21, 1992. [9205120277]

[32] Memorandum for J. Taylor from T. Murley, "Planned Actions to Address the Issues from the Office of Inspector General"s Report on the NRC Staff"s Review and Acceptance of Thermo-Lag 330-1 Fire Barrier Material," August 21, 1992. [9209250288]

[33] Memorandum for C. Serpan from A. Chaffee, "Nuclear Reactor Regulation (NRR) Input Into Research NUREG-0933 (WITS Item 9400213)," February 13, 1996. [9602260124]

[34] Bulletin 92-01, "Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free from Fire Damage," U.S. Nuclear Regulatory Commission, June 24, 1992. [ML031250239]

[35] Memorandum for J. Taylor from T. Murley, "Planned Actions to Address the Issues from the Office of Inspector General"s Report on the NRC Staff"s Review and Acceptance of Thermo-Lag 330-1 Fire Barrier Material," August 21, 1992. [9209250288]

requiring compliance with existing NRC requirements and to permit plant-specific exemptions, where justified. As of December 1995, the staff had issued several generic communications regarding Thermo-Lag fire barriers, including INs 91-47,[36] 91-79,[37] 92-55,[38] 92-82,[39] 94-22,[40] 94-34,[41] 94-86,[42] 95-27,[43] 95-32,[44] and 95-49.[45] Two major items of the action plan remain to be completed: (1) mechanical properties tests; and plant-specific fire test curve feasibility study.

On 10/03/95, NEI submitted to the NRC the NUCON International, Inc. Report 06VA764/04, "Pyrolysis Gas Chromatography Analysis and Energy Dispersive Spectroscopy of Thermo-Lag Fire Barrier Samples." In its letter to the NRC, NEI stated that, on the basis of the tests, all samples (169 from 18 utilities representing 25 nuclear power plants) contained the constituents essential to fire barrier performance and the composition of the samples was consistent. The staff performed chemical composition tests and analyses at NIST which confirmed the results of the NEI analyses.

Concerns about the reliability of information and data supplied by TSI prompted the staff to reassess previous technical conclusions and determine the extent to which the NRC or the nuclear industry relied on information supplied by TSI to reach these conclusions. The staff identified and categorized the issues and previous conclusions and, on the basis of the results of the chemical analysis performed by NIST and NEI, concluded that additional action was not needed to reassess the issues or verify the conclusions. The staff continues to work with NIST to evaluate the feasibility of developing fire curves for rating fire barriers on the basis of representative nuclear power plant fire hazards rather than the fire curves specified in existing fire test standards.

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[36] Information Notice 91-47, "Failure of Thermo-Lag Fire Barrier Material to Pass Fire Endurance Test," U.S. Nuclear Regulatory Commission, August 6, 1991. [ML031190452]

[37] Information Notice 91-79, "Deficiencies in the Procedures for Installing Thermo-Lag Fire Barrier Materials," U.S. Nuclear Regulatory Commission, December 6, 1991 [ML031190449], (Supplement 1) August 4, 1994. [ML070180066]

[38] Information Notice 92-55, "Current Fire Endurance Test Results for Thermo-Lag Fire Barrier Material,"U.S. Nuclear Regulatory Commission, July 27, 1992. [ML031200114]

[39] Information Notice 92-82, "Results of Thermo-Lag 330-1 Combustibility Testing," U.S. Nuclear Regulatory Commission, December 15, 1992. [9212090211]

[40] Information Notice 94-22, "Fire Endurance and Ampacity Derating Test Results for 3-Hour Fire-Rated Thermo-Lag 330-1 Fire Barriers," U.S. Nuclear Regulatory Commission, March 16, 1994. [ML031060605]

[41] Information Notice 94-34, "Thermo-Lag 330-660 Flexi-Blanket Ampacity Derating Concerns," U.S. Nuclear Regulatory Commission, May 13, 1994. [ML031060554]

[42] Information Notice 94-86, "Legal Actions Against Thermal Science, Inc., Manufacturer of Thermo-Lag," U.S. Nuclear Regulatory Commission, December 22, 1994. [ML031060390]

[43] Information Notice 95-27, "NRC Review of Nuclear Energy Institute, "Thermo-Lag 330-1 Combustibility Evaluation Methodology Plant Screening Guide," U.S. Nuclear Regulatory Commission, May 31, 1995. [ML031060160]

[44] Information Notice 95-32, "Thermo-Lag 330-1 Flame Spread Test Results," U.S. Nuclear Regulatory Commission, August 10, 1995. [ML031060314]

[45] Information Notice 95-49, "Seismic Adequacy of Thermo-Lag Panels," U.S. Nuclear Regulatory Commission, October 27, 1995. [ML031210507]

GA-005: RCS DRAINDOWN DESCRIPTION

On 09/17/94, Wolf Creek experienced a loss of RCS inventory while transitioning to a refueling shutdown. The event occurred when operators cycled a valve in the Train A side of the RHR system cross-connect line following maintenance on the valve, while at the same time establishing a flow path from the RHR system, Train B, to the refueling water storage tank for reborating train B. The failure of the reactor operating staff to adequately control two incompatible activities resulted in transferring 9,200 gallons of hot RCS water to the RWST in 66 seconds. A study of the event was documented in AEOD/S95-01.[46] This activity was identified in an NRR memorandum[47] to RES in February 1996.

The Wolf Creek event represents a LOCA with the potential to consequentially fail all the ECCS pumps and bypass the containment. Another important feature of this event is the short time available for corrective action. Based upon calculations by the licensee and the staff, it is estimated that if the draindown had not been isolated within 3 to 5 minutes, NPSH would have been lost for all ECCS pumps, and core uncovery would have followed in about 25 to 30 minutes. This event represented a vulnerability in PWRs that was not previously recognized.

An Action Plan[48] was developed to collect and evaluate information from licensees regarding plant-specific system configurations and vulnerabilities to draindown events. Specific actions included in the plan were staff issuance of: (1) an IN to alert licensees to the Wolf Creek event; and (2) a GL requesting all PWR licensees to provide information on draindown vulnerabilities and the measures that have been implemented to diminish the probability of a draindown.

CONCLUSION

The staff performed an evaluation of the probability of event initiation and of the conditional core damage probability. The resultant low value of the core damage probability along with licensee awareness of the failure

scenario made the risk from the Wolf Creek event small and IN 95-03[49] was issued.

GA-006: SRP REVISION DESCRIPTION

The SRP Update and Development Program (SRP-UDP) was established[50][51] [52] in 1991 to update the

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[46] AEOD/S95-01, "Reactor Coolant System Blowdown at Wolf Creek on September 17, 1994," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, March 1995. [9503310036]
[47] Memorandum for C. Serpan from A. Chaffee, "Nuclear Reactor Regulation (NRR) Input Into Research NUREG-0933 (WITS Item 9400213)," February 13, 1996. [9602260124]
[48] Memorandum for A. Thadani from R. Jones, "Proposed Action Plan for the 'Wolf Creek Draindown Event,'" September 8, 1995. [9509140225]
[49] Information Notice 95-03, "Loss of Reactor Coolant Inventory and Potential Loss of Emergency Mitigation Functions While in a Shutdown Condition," U.S. Nuclear Regulatory Commission, January 18, 1995 [ML031060404], (Supplement 1) March 25, 1996. [ML082840801]
[50] Memorandum for the Chairman from J. Taylor, "Commercial Contract for Technical Assistance to Support the Standard Review Plan Update and Development Program," November 18, 1991.
[51] Memorandum for the Chairman from J. Taylor, "Commercial Contract for Technical Assistance to Support the Standard Review Plan Update and Development Program," November 18, 1991.
[52] Memorandum for J. Taylor from I. Selin, "Commercial Contract for Technical Assistance to Support the Standard Review Plan Update and Development Program," December 13, 1991.

SRP[53] for use in reviewing future reactor design applications. The revised SRP[54] incorporates changes in the regulation of the nuclear power industry that have occurred since the last SRP revision in 1981. In SECY-91-161,[55] the staff discussed, in part, the revision effort for the SRP and committed to produce supplements to the 1981 SRP in parallel with the conduct of future reactor design reviews.

The SRP Revision Action Plan[56] deals with the development of draft revisions for all sections, except Chapter 7, and the development of new SRP sections to cover review areas that are supported by established staff positions or are fully addressed in the evolutionary reactor design reviews. The draft revisions will incorporate recommended changes identified in the review of generic regulatory documents and NRR staff SERs for evolutionary LWR designs. Specific tasks included in the Action Plan are:

(1) Identify established staff positions and new regulatory requirements from a review of generic regulatory documents issued since the last SRP revision and from a review of NRR staff safety evaluation reports for evolutionary LWR designs;

(2) Prepare a side-by-side comparison of the SRP-cited version of codes and standards vs. the current version of the standard;

(3) Prepare draft revisions of the current SRP sections to incorporate the changes recommended;

(4) Prepare new draft SRP sections that are supported by established staff positions or are fully addressed in the evolutionary design reviews;

(5) Automate the SRP to make future revisions and accessibility easier to accomplish;

(6) Maintain the program data base to reflect new staff positions and requirements.

This activity was identified in an NRR memorandum[57] to RES in February 1996.

CONCLUSION

NRR has established the SRP-UDP to update the SRP for use in the review of future reactor applications to reflect existing agency requirements and guidance and to add new review criteria to accommodate future designs. An automated version of the current SRP has been developed and is operational.

GA-007: PRA IMPLEMENTATION PLAN

The NRC has been making use of PRA technology to varying degrees in its regulatory activities since the issuance of WASH-1400.[58] Prior to 1991, this had been an ad hoc application, depending on the availability of expertise in various technical groups. Since 1991, there have been a number of high level studies within NRC that focused on the status of PRA use and its role in the regulatory process. Collectively, the findings

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[53] NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (1st Ed.) November 1975, (2nd Ed.) March 1980, (3rd Ed.) July 1981.

[54] NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (1st Ed.) November 1975, (2nd Ed.) March 1980, (3rd Ed.) July 1981.

[55] SECY-91-161, "Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions," U.S. Nuclear Regulatory Commission, May 31, 1991. [9106050174]

[56] Memorandum for W. Russell from F. Gillespie, "Action Plan for the Development of Draft SRP Revisions in the SRP-UDP," May 17, 1994. [9406280148, 9405270273]

[57] Memorandum for C. Serpan from A. Chaffee, "Nuclear Reactor Regulation (NRR) Input Into Research NUREG-0933 (WITS Item 9400213)," February 13, 1996. [9602260124]

[58] WASH-1400 (NUREG-75/014), "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Atomic Energy Commission, October 1975.

and recommendations from these studies support the view that there is a need for increased emphasis on PRA technology applications.

For the full value of the NRC'S investment in risk assessment methodology to be achieved, it is important that consistent high-level agency guidance be provided on the appropriate use of PRA. To this end, in November 1993, the Office Directors of NRR, AEOD, NMSS, and RES proposed to take the initiative in providing guidance on coordination and expectations for PRA efforts. Specifically, they proposed to develop an integrated plan

for the staff's risk assessment and risk management practices.[59] In August 1994, the staff submitted SECY-94-219[60] for the Commission's information. On 03/30/95, the staff submitted SECY-95-079[61] and

briefed the Commission on the subject on 04/05/95. On 05/18/95, the staff forwarded SECY-95-126[62] to the Commission and a final PRA Policy Statement was published in the Federal Register on 08/16/95.

An Action Plan[63] was developed to describe the process for the staff to use PRA methods and technology in the NRC's effort toward a risk-informed regulatory approach; this plan encompasses methods development, pilot applications, and staff training. The plan will be used to ensure timely and integrated agency-wide effort that is consistent with the PRA Policy Statement and is meant to improve the regulatory process by developing state- of-the-art PRA tools that will expand the use of PRA technologies in making regulatory decisions; the plan is not intended to correct safety problems at licensed facilities.

The PRA Implementation Plan includes activities for NRR, RES, AEOD, and NMSS staff to increase the use of PRA methods in all regulatory matters. NRR focuses on the PRA applications in reactor regulations, the development of standard review plans, the pilot programs to use PRA technology in specific regulatory

initiatives, events assessment, and working with regions on risk-informed inspections. RES focuses on the IPE/ IPEEE reviews, PRA method and quality, and the development of PRA regulatory guides for the industry. AEOD focuses on risk-informed trends and patterns analysis, reliability data for PRA applications, and staff training.

NMSS focuses on using PRA in high and low level waste issues. The detailed actions are described in the PRA Implementation Plan.

On 11/17/95, a memorandum was forwarded to senior NRR management providing additional guidance on implementing the Commission's PRA Policy Statement and managing tasks contained in the PRA Implementation Plan. As a result of this memorandum, several additional action plans were expected to be developed for individual line items in the PRA Implementation Plan. In addition, more detailed information concerning PRA Implementation Plan activities will be collected so that more accurate and timely status

of all activities can be maintained in the ongoing PRA Implementation Plan. On 11/27/95, the staff issued

SECY-95-280[64] to provide a general structure to ensure consistent and appropriate application of PRA methods and to outline a process for developing guidance and standards. On 11/30/95, Chairman Jackson issued a memorandum requesting the staff to develop action plans and timetables to provide better focus and accelerate NRC's risk-informed regulatory effort. In response to this request, the EDO forwarded a

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[59] Memorandum for J. Taylor from T. Murley et al. "Agency Directions for Current and Future Uses of Probabilistic Risk Assessment (PRA)," November 2, 1993. [9311100145]
[60] SECY-94-219, "Proposed Agency-Wide Implementation Plan for Probabilistic Risk Assessment (PRA)," U.S. Nuclear Regulatory Commission, August 19, 1994. [9409090234]
[61] SECY-95-079, "Status Update of the Agency- Wide Implementation Plan for Probabilistic Risk Assessment," U.S. Nuclear Regulatory Commission, March 30, 1995. [9504100180]
[62] SECY-95-126, "Final Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," U.S. Nuclear Regulatory Commission, May 18, 1995. [9506020152]
[63] SECY-94-219, "Proposed Agency-Wide Implementation Plan for Probabilistic Risk Assessment (PRA)," U.S. Nuclear Regulatory Commission, August 19, 1994. [9409090234]
[64] SECY-95-280, "Framework for Applying Probabilistic Risk Analysis in Reactor Regulation," U.S. Nuclear Regulatory Commission, November 27, 1995. [9512180168, 9512040133]

memorandum[65] to Chairman Jackson on 01/03/96 which described the staff action plan for utilizing PRA in reactor-related activities, including the PRA pilot programs and the accelerated milestones for the development of regulatory guidance documents.

RGA-007.1.2(D): GRADED QUALITY ASSURANCE DESCRIPTION

This task called for the preparation of staff evaluation guidance and regulatory guidance for industry implementation for the grading of QA practices commensurate with the safety significance of the plant equipment. The development of this guidance will be based on staff reviews of regulatory requirements, proposed changes to existing practices, and assessment of the actual programs developed by the three volunteer utilities implementing graded QA programs.

10 CFR 50 Appendices A and B require QA programs that are commensurate (or consistent) with the importance to safety of the functions to be performed. However, the QA implementation practices that have evolved have often not been graded. In the development of implementation guidance for the maintenance rule, a methodology to determine the risk significance of plant equipment was proposed by the industry (NUMARC 93-01). During a public meeting on 12/16/93 the staff suggested that the industry could build on the experience gained from the maintenance rule to develop implementation methodologies for graded QA. The staff had numerous interactions with NEI during 1994 as the graded QA concepts were discussed and the initial industry guidelines were developed and commented on. In early 1995, the licensees of Grand Gulf, South Texas, and Palo Verde volunteered to work with the staff. The staff has reviewed the licensee developmental graded QA

efforts. This activity was identified in an NRR memorandum[66] to RES in February 1996.

The goal of the Action Plan[67] is to utilize the lessons learned from the 3 volunteer licensees to modify staff- developed draft guidance to formulate regulatory guidance on acceptable methods for implementing graded QA. The staff will develop a regulatory guide, a revision to Chapter 17 of the SRP, and a reactive inspection procedure (IP) for graded QA. An inter-office team will be established to prepare the regulatory guidance documents and test their implementation during the evaluation of volunteer plant activities.

Existing regulations provide the necessary flexibility for the development and implementation of graded QA programs. The staff will issue a NUREG report regarding the lessons learned from the volunteer plant implementations. Additional regulatory guidance will be issued to either disseminate staff guidance or

endorse an industry approach. Planned guidance for the staff will involve an evaluation guide for application to the volunteer plants, the lessons learned report, training sessions and public workshops, SRP revisions, and inspection guidance in the form of a reactive IP. The staff is evaluating the appropriate mechanism for inspections of the risk significance determination aspects of graded QA programs.

The safety benefits to be gained from a graded QA program could be significant since both NRC reviews and inspections and the industry's quality control resources would be focused on the more safety significant plant equipment and activities. Secondarily, cost savings to the industry could be realized by avoiding the dilution of resources expended on less safety significant issues.

CONCLUSION

A draft evaluation guide for NRC use has been prepared for application to the volunteer plants implementing graded QA programs. The staff will utilize the guide for the review of the volunteer plant graded QA programs. The guide and the staff's proposed interaction framework were expected to be

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[65] Memorandum for Chairman Jackson from J. Taylor, "Improvements Associated with Managing the Utilization of Probabilistic Risk Assessment (PRA) and Digital Instrumentation and Control Technology," January 3, 1996. [9601180203]

[66] Memorandum for C. Serpan from A. Chaffee, "Nuclear Reactor Regulation (NRR) Input Into Research NUREG-0933 (WITS Item 9400213)," February 13, 1996. [9602260124]

[67] SECY-94-219, "Proposed Agency-Wide Implementation Plan for Probabilistic Risk Assessment (PRA)," U.S. Nuclear Regulatory Commission, August 19, 1994. [9409090234]

transmitted to the three volunteer licensees for comments.