UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 25, 1990

TO: ALL HOLDERS OF OPERATING LICENSES AND CONSTRUCTION PERMITS FOR NUCLEAR POWER REACTORS

SUBJECT: REQUEST FOR INFORMATION ON THE STATUS OF LICENSEE IMPLEMENTATION OF GENERIC SAFETY ISSUES RESOLVED WITH IMPOSITION OF REQUIREMENTS OR CORRECTIVE ACTIONS (GENERIC LETTER 90-04)

This letter is being issued as part of our continuing effort to establish and maintain an accurate and validated implementation status for all significant staff-imposed regulatory requirements or corrective actions. It requests that you review and provide documentation of the current implementation status of all generic safety issues (GSIs) identified herein that apply to your facility. An important objective of this effort is to obtain licensee and staff agreement on the GSI implementation status at each reactor facility.

A GSI is a safety concern, as identified and characterized in NUREG-0933, "A Prioritization of Generic Safety Issues," that affects the design, construction, or operation of all, several, or a class of nuclear power plants and may have the potential for safety improvements at such plants. This request applies to those GSIs which have been resolved by the staff and whose resolutions have involved the promulgation of new or revised requirements or guidance to the industry. The determination of the status of other generic activities, such as multiplant actions (MPAs) not designated as GSIs, that have imposed requirements on or requested action of licensees is not included in this request, but is being tracked separately.

Enclosure 1 is a table of GSIs that we have included in this request. We have provided the GSI number and associated MPA number, where applicable, the GSI title and the applicability of the issue to various classes of facilities. You should complete the "Status" column in accordance with the guidance that accompanies Enclosure 1. To assist you, we have provided a summary of each GSI and its resolution in Enclosure 2, along with applicable documentation references.

As in our previous requests related to the implementation status of the TMI Action Plan items (individual letters to licensees, April 1989) and the unresolved safety issues (USIs), (GL 89-21, October 1989), implementation is considered complete when you have performed all of the actions necessary to satisfy the requirements, corrective actions, or assumptions in the staff's technical resolution of the GSI.

We request that you respond to this letter by June 29, 1990. In preparing your response we suggest that you coordinate with your NRC Project Manager to resolve any questions.

This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires on January 31, 1991. The estimated average number of burden hours is 80 person-hours per facility, including searching data sources, gathering the data, and preparing the required response. These estimated average burden hours pertain only to the identified response-related matters and do not include implementation of the recommendations or requirements that resulted from resolution of the GSIs. Send comments regarding this burden estimate or any other aspect of this collection of information, including

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suggestions for reducing this burden, to the Information and Records Management Branch, Division of Information Support Services, Office of Information Resources Management (MNBB-7714), U.S. Nuclear Regulatory Commission, Washington, D.C. 20555; and to the Paperwork Reduction Project (3150-0011), Office of Management and Budget, Washington, D.C. 20503.

Please address your response to this generic letter to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, pursuant to 10 CFR Section 50.4 of the NRC's regulations.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. GSI Table
2. GSI Summaries
3. List of Most Recently Issued NRC Generic Letters

Enclosure 1

Status of Licensee Implementation of
Generic Safety Issues Resolved With
Imposition of Requirements or Corrective Actions

FACILITY NAME: _____
DOCKET NO.: _____
LICENSEE: _____

STATUS OF LICENSEE IMPLEMENTATION OF GENERIC SAFETY ISSUES

RESOLVED WITH IMPOSITION OF REQUIREMENTS OR CORRECTIVE ACTIONS

GSI/(MPA No.)	TITLE	APPLICABILITY	STATUS*	COMMENTS
40 (B065)	Safety Concerns Associated With Pipe Breaks In The BWR Scram System	All BWRs		
41 (B058)	BWR Scram Discharge Volume Systems	All BWRs		
43 (B107)	Reliability Of Air Systems	All Plants		
51 (L913)	Improving the Reliability of Open-Cycle Service Water Systems	All Plants		
67.3.3 (A017)	Improved Accident Monitoring	All Plants		
75** (B076)	Item 1.1 - Post-Trip Review (Program Description and Procedure)	All Plants		
75 (B085)	Item 1.2 - Post-Trip Review - Data and Information Capability	All Plants		

*Please follow attached guidance for completing this column.

**The 16 items listed for GSI 75 all relate to actions derived from the generic implications of Salem ATWS events. Item numbers correspond to Generic Letter 83-28 action item numbers.

GSI/(MPA No.)	TITLE	APPLICABILITY	STATUS*	COMMENTS
75 (B077)	Item 2.1 - Equipment Classification and Vendor Interface (Reactor Trip System Components)	All Plants		
75 (B086)	Item 2.2.1 - Equipment Classification for Safety-Related Components	All Plants		
75 (L003)	Item 2.2.2 - Vendor Interface for Safety-Related Components	All Plants		
75 (B078)	Items 3.1.1 & 3.1.2 - Post - Maintenance Testing (Reactor Trip System Components)	All Plants		
75 (B079)	Item 3.1.3 - Post-Maintenance Testing-Changes to Test Requirements (Reactor Trip System Components)	All Plants		
75 (B087)	Items 3.2.1 & 3.2.2 - Post-Maintenance Testing (All Other Safety-Related Components)	All Plants		
75 (B088)	Item 3.2.3 - Post-Maintenance Testing-Changes to Test Requirements (All Other Safety-Related Components)	All Plants		
75 (B080)	Item 4.1 - Reactor Trip System Reliability (Vendor-Related Modifications)	All Plants		

GSI/(MPA No.)	TITLE	APPLICABILITY	STATUS*	COMMENTS
75 (B081)	Items 4.2.1 & 4.2.2 - Reactor Trip System Reliability-Maintenance and Testing (Preventative Maintenance and Surveillance Program for Reactor Trip Breakers)	All PWRs		
75 (B082)	Item 4.3 - Reactor Trip System Reliability - Design Modifications (Automatic Actuation of Shunt Trip Attachment for Westinghouse and B&W Plants)	All W and B&W Plants		
75 (B090)	Item 4.3 - Reactor Trip System Reliability - Tech Spec Changes (Automatic Actuation of Shunt Trip Attachment For Westinghouse and B&W Plants)	All W & B&W Plants		
75 (B091)	Item 4.4 - Reactor Trip System Reliability (Improvements in Maintenance and Test Procedures for B&W Plants)	All B&W Plants		

GSI/(MPA No.)	TITLE	APPLICABILITY	STATUS*	COMMENTS
75 (B092)	Item 4.5.1 - Reactor Trip System Reliability-Diverse Trip Features (System Functional Testing)	All Plants		
75 (B093)	Items 4.5.2 & 4.5.3 - Reactor Trip System Reliability - Test Alternatives and Intervals (System Functional Testing)	All Plants		
86 (B084)	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	All BWRs		
93 (B098)	Steam Binding of Auxiliary Feedwater Pumps	All PWRs		
99 (L817)	RCS/RHR Suction Line Valve Interlock on PWRs	All PWRs		
124	Auxiliary Feedwater System Reliability	ANO-1&2, Rancho Seco, Prairie Island 1&2, Crystal River-3 Ft. Calhoun		
A-13 (B017)	Snubber Operability Assurance - Hydraulic Snubbers	All Plants		

GSI/(MPA No.)	TITLE	APPLICABILITY	STATUS*	COMMENTS
A-13 (B022)	Snubber Operability Assurance - Mechanical Snubbers	All Plants		
A-16 (D012)	Steam Effects on BWR Core Spray Distribution	Oyster Creek & NMP-1		
A-35 (B023)	Adequacy of Offsite Power Systems	All Plants		
B-10	Behavior of BWR Mark III Containments	All BWR Mark III Plants		
B-36	Develop Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and for Normal Ventilation Systems	All Plants with OL Applications After 4/1/80		
B-63 (B045)	Isolation of Low Pressure Systems Connected to the Reactor Coolant System Pressure Boundary	All Plants		

Attachment to
Enclosure 1

Guidance For Completing Status Column in Enclosure 1

(1) Provide a separate entry for each licensed reactor unit. If the information is identical for multiple units, so state.

(2) If a GSI is not applicable to a unit(s), enter "NA".

(3) If a GSI is applicable but no changes were necessary to implement the resolution, enter "NC". If the GSI implementation was completed prior to issuance of the operating license, enter "NC", as no post-licensing changes were necessary.

(4) If a GSI is applicable, submittal of information and/or changes were necessary and such submittals were made or changes are complete, enter "C". Also identify the licensee's document(s) to the NRC which certified completion, and the document date(s).

(5) If a GSI is applicable and changes are necessary but such changes are not yet fully implemented, enter "I" and the projected month and year of completion. Provide a brief explanation of the outstanding work in the "Comments" column.

(6) If implementation guidance for a resolved GSI was issued recently and the licensee is still evaluating the appropriate response, enter "E" and the projected response date.

(7) The "Comments" column may be used to explain any entry in the "Status" column.

Enclosure 2

Generic Safety Issue Summaries

NOTE: For further details on any of the issues, consult NUREG-0933

GSI No. 40 (MPA No. B-065) TITLE: Safety Concerns Associated With Pipe Breaks in the BWR Scram System

This issue arose from staff concerns related to the possibility of a break or leakage in scram discharge volume (SDV) piping which could environmentally threaten safety-related equipment, or introduce problems in maintaining reactor coolant system inventory.

On April 10, 1981, the NRC staff sent a generic letter to all BWR applicants and licensees requesting them to provide their plant-specific responses addressing the concerns identified in NUREG-0785. Subsequently, Generic Letters 81-34 and 81-35 were sent to BWR licensees and applicants, respectively, wherein it was stated that plant-specific responses conforming to the guidance contained in NUREG-0803 would satisfy the request for information in the April 10, 1981 letter.

The staff's resulting generic Safety Evaluation Report for this issue was transmitted to all BWR applicants and licensees by Generic Letter 86-01. The evaluation concluded that through-wall cracks in the SDV piping need not be postulated. In addition, even if a through-wall flaw is initially present in the SDV system, it will not propagate into a break under the staff-defined piping loads. Further, leakage from such a flaw will be small and, therefore, a harsh environment over large areas of the reactor building which could affect redundant safety-related mitigating equipment will not result. Thus, the potentially exposed safety-related equipment need not be qualified for operation in a harsh environment associated with an SDV break.

References:

1. NUREG-0785, "Safety Concerns Associated With Pipe Breaks in the BWR Scram System," U.S. Nuclear Regulatory Commission, April 1981.

2. Letter to All BWR Licensees from D. Eisenhut, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System," April 10, 1981.

3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," U.S. Nuclear Regulatory Commission, August 1981.

4. Letter to All GE BWR Licensees (Except Humboldt Bay) from D. Eisenhut, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System (Generic Letter 81-34)," August 31, 1981.

5. Letter to All BWR Applicants for CPs, Holders of CPs, and Applicants for OLs from D. Eisenhut, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System (Generic Letter 81-35)," August 31, 1981.

6. Letter to All BWR Applicants and Licensees, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System (Generic Letter 86-01)," January 3, 1984.

GSI No. 41 (MPA No. B-058) TITLE: BWR Scram Discharge Volume Systems

This issue arose from staff concerns related to the Browns Ferry 3 partial scram failure event of June 28, 1980, failures of scram level instruments, and subsequent staff evaluations of boiling water reactor (BWR) scram discharge volume (SDV) systems.

The staff's resulting generic Safety Evaluation Report of SDV systems was transmitted to all BWR licensees and applicants by letter dated December 9, 1980. This letter identified both short- and long-term corrective action programs. The short-term actions were covered by Bulletins 80-14 and 80-17, as supplemented. GSI No. 41 addressed the long-term program.

The resolution of this GSI affected all BWRs and addressed the following long-term actions: (1) improvement of the hydraulic coupling between the SDV headers and the instrumented volume; (2) improvement of the reliability of the float switches in the instrumented volume; (3) modification of the instrumented volume to prevent level sensor damage from hydrodynamic forces and water hammer during a scram; and (4) submittal of Technical Specifications changes appropriate to the modified SDV systems.

A BWR Owner's Group developed criteria to implement the resolution and the criteria were endorsed by the staff with addition by the staff of a criterion for diverse level instrumentation. Licensee commitments to implement the permanent corrective actions were confirmed by NRC orders issued in June 1983.

References:

1. Letter to All BWR Licensees, "BWR Scram Discharge System," December 9, 1980.

2. IE Bulletin No. 80-14, "Degradation of BWR Scram Discharge Volume Capability," June 12, 1980.

3. IE Bulletin No. 80-17, "Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR," July 3, 1980.

4. IE Bulletin No. 80-17, Supplement 1, July 18, 1980.

5. IE Bulletin No. 80-17, Supplement 2, "Failures Revealed by Testing Subsequent to Failure of Control Rods to Insert During a Scram at a BWR," July 22, 1980.

GSI No. 43 (MPA No. B-107) TITLE: Reliability of Air Systems

This issue arose from staff concerns related to the Three Mile Island accident and subsequent air-operated equipment failures at other plants. Some of these

equipment failures are described in Information Notice (IN) 87-28 and IN 87-28, Supplement 1.

The staff's generic Safety Evaluation Report, NUREG-1275, V.2, was provided to all licensees and applicants by IN 87-28, Supplement 1. Generic Letter 88-14 identified requested corrective actions. These actions consisted of three types of verification and a discussion of a program for maintaining air quality. The three types of verification included: (1) test verification of air quality, (2) verification of adequate maintenance practices, emergency procedures, and training, and (3) verification of design and failure modes. Responses concerning implementation of these actions were to be submitted within 180 days with allowances made for implementation of actions requiring outages to complete.

References:

1. NRC Information Notice No. 87-28, "Air Systems Problems at U.S. Light Water Reactors," June 22, 1987.

2. NRC Information Notice No. 87-28, Supplement 1, December 28, 1987.

3. NUREG-1275, "Operating Experience Feedback Report - Air Systems Problems," U.S. Nuclear Regulatory Commission, Vol. 2, December 1987.

4. NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants, "Instrument Air Supply Systems Problems Affecting Safety-Related Equipment (Generic Letter 88-14)," August 8, 1988.

GSI No. 51 (MPA No. L-913) TITLE: Improving the Reliability of Open-Cycle Service Water Systems

This issue arose from operating experience and studies related to Bulletin 81-03 which led the NRC to question the compliance of the service water systems with the requirements of GDC 44, 45, 46 and Appendix B to 10 CFR Part 50.

The resolution of GSI No. 51, along with implementation of AEOD and Region II recommendations, affected all plants and addressed the following actions: (1) reduce flow blockage problems from biofouling, (2) conduct a heat transfer testing program on safety-related heat exchangers in open-cycle systems, (3) establish a routine inspection and maintenance program for open-cycle system piping and components, (4) confirm that the service water system will perform its intended function in accordance with the licensing basis for the plant; and (5) confirm the adequacy of relevant maintenance practices, operating and emergency procedures, and training.

Generic Letter 89-13 requested licensees to advise the staff whether they have established programs to implement the above five actions resulting from the resolution of GSI No. 51, or equally effective alternative courses of action. The Generic Letter also requested licensees to confirm to the staff that all recommended actions or equivalent alternatives have been implemented.

References:

1. NRC Bulletin No. 81-03, "Flow Blockage of Cooling Water to Safety System Components by Corbicula sp. (Asiatic Clam) and Mytilus sp. (Mussel), " April 10, 1981.

2. NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants, "Service Water System Problems Affecting Safety-Related Equipment (Generic Letter 89-13)," July 18, 1989.

GSI No. 67.3.3 (MPA A-017) TITLE: Improved Accident Monitoring

This issue addresses compliance with Regulatory Guide 1.97. NUREG-0737, "Clarification of TMI Action Plan Requirements," was issued in 1980, followed by Supplement 1 (issued as Generic Letter 82-33) in December 1982. Supplement 1 requested proposed schedules for implementing the provisions of Revision 2 to Regulatory Guide 1.97. In addition, licensees and applicants were requested to submit details, for staff review, of how they would comply with the provisions of Regulatory Guide 1.97, Rev. 2, and to identify any exceptions to or deviations from these provisions.

Based on licensee responses to Supplement 1, confirmatory orders were issued to operating plants in 1985. For license applications still under review, implementation would be addressed as part of the licensing process.

References:

1. Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," U.S. Nuclear Regulatory Commission, December 1980.

2. NRC Letter to Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, "Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability (Generic Letter No. 82-33)," December 17, 1982.

GSI No. 75 (17 Individual MPAs) TITLE: Generic Implications of ATWS Events at the Salem Nuclear Plant

This issue arose from staff concerns resulting from analysis of events that occurred at the Salem Nuclear Power Plant on February 22 and 25, 1983. The analysis of the events revealed that a total loss of automatic scram capability (an anticipated transient without scram, or ATWS event) had occurred each time. The relatively mild transients, coupled with the rapid manual shutdown of the reactor by the operators both times, turned these potentially serious events into little more than routine reactor shutdowns. However, the implications of these events vis a vis scram system reliability were considered to be extremely safety-significant.

The study of these events resulted in the issuance of NUREG-1000 and Generic Letter 83-28. The Generic Letter contained a number of items and sub-items addressing those aspects of GSI 75 which have been resolved, each requesting specified actions of all or identified categories of licensees and applicants.

It should be noted that two aspects of GSI 75 have not yet been fully resolved and thus are not included herein. One of these was not addressed in GL 83-28 and involves possible revisions to Reg. Guide 1.33, "QA Program Requirements (Operations)" to contain more detailed guidance for operational QA programs. The second relates to Items 4.2.3 and 4.2.4 of GL 83-28 which address life testing and replacement of reactor trip system components. The staff is currently reassessing the methods for establishing reactor trip reliability and may issue a future generic communication on these items.

The 16 sub-issues of GSI 75, described below, consist of items and sub-items from GL 83-28 in accordance with how they were grouped into Multi-plant Actions (MPAs) by the staff for tracking purposes. Each sub-issue relates to a single MPA and may contain more than one sub-item from GL 83-28.

References:

1. NRC Letter to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, "Required Actions Based on Generic Implications of Salem ATWS Events (Generic Letter 83-28)," July 8, 1983.

2. NUREG-1000, Volume 1, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant," U.S. Nuclear Regulatory Commission, April 1983.

3. NUREG-1000, Volume 2, August 1983.

(MPA No. B-076) TITLE: Item 1.1- Post-Trip Review (Program Description and Procedure)

The resolution of this item, applicable to all plants, requests that licensees and applicants describe their programs for ensuring that unscheduled reactor shutdowns are analyzed and a determination made that the plant can be restarted safely.

As a minimum, each licensee is requested to describe: (1) the criteria for determining the acceptability of restart, (2) the responsibilities and authorities of personnel who perform the review and analysis, (3) the necessary qualifications and training for the responsible personnel, (4) the sources of plant information necessary to conduct the review and analysis, (5) the methods and criteria for comparing the event information with known or expected plant behavior, (6) the criteria for determining the need for an independent assessment of an event and guidelines on the preservation of physical evidence to support independent analysis of the event, and (7) the systematic safety assessment procedures compiled from (1) to (6) which are used in conducting the evaluation by the staff.

(MPA No. B-085) TITLE: Item 1.2 - Post-Trip Review - Data and Information Capability

Item 1.2 requests that licensees and applicants have the capability to record, recall, and display data and information to permit diagnosing the causes of unscheduled reactor shutdowns and the proper functioning of safety-related equipment during these events using systematic safety assessment procedures. The data and information are to be displayed in a form that is user-friendly and reflects human factors considerations. It further requests licensees and applicants to prepare and submit a report which describes and justifies the adequacy of their equipment for diagnosing an unscheduled reactor shutdown. Submittals are to be reviewed by the staff to determine whether adequate data and information will be available to support the systematic assessment of unscheduled reactor shutdowns.

(MPA No. B-077) TITLE: Item 2.1 - Equipment Classification and Vendor Interface (Reactor Trip System Components)

Item 2.1 addresses components whose functioning is required to trip the reactor and requests all licensees and applicants to describe their program to assure that all such components are identified as "safety-related" in documents, procedures and information handling systems used to control safety-related activities in the plant. In addition, the item requests that a vendor interface program be established, implemented and maintained for such components to ensure that relevant vendor information is complete, current and controlled throughout the plant lifetime, that it is appropriately referenced or incorporated in plant instructions and procedures, and that it include periodic communication with the vendor. The licensees' submittals are to be reviewed by the staff to determine their adequacy.

(MPA No. B-086) TITLE: Item 2.2.1 - Equipment Classification for Safety-Related Components)

Item 2.2.1 addresses all other safety-related components and requests all licensees and applicants to describe their program used to classify such components. The classification program is necessary to ensure that all such components are identified as "safety-related" in documents, procedures and

information handling systems used to control safety-related activities in the plant, and must include periodic communication with the vendor. The staff is to review the licensees' submittals to determine their adequacy.

This MPA originally addressed vendor interface programs for safety-related components in addition to component classification, as identified in GL 83-28. The original vendor interface program guidelines were modified and superseded by way of GL 90-03 on March 20, 1990. A new MPA was established to track implementation of the revised guidelines. They are discussed separately below.

Additional Reference:

1. NRC Letter to All Power Reactor Licensees and Applicants, "Relaxation of Staff Position in Generic Letter 83-28, Item 2.2 Part 2 'Vendor Interface for Safety-Related Components' (Generic Letter No. 90-03)," March 20, 1990.

(MPA No. L-003) TITLE: Item 2.2.2 - Vendor Interface for Safety-Related Components

The original needs for vendor interface programs for safety-related components were specified in GL 83-23 and licensee implementation was being tracked via MPA No. B-086, together with equipment classification. GL 90-03 was issued on March 20, 1990 which relaxes and supersedes the original vendor interface program guidance based upon industry initiatives and experience. The revised interface program with the NSSS vendor covers all safety-related components within the NSSS scope of supply and is to conform with the Vendor Equipment Technical Information Program (VETIP) as described in the Nuclear Utility Task Action Committee Report, INPO 84-010 issued in March 1984. A program of periodic contact with non-NSSS vendors of other key safety-related components is also specified.

Additional References:

1. NRC Letter to All Power Reactor Licensees and Applicants, "Relaxation of Staff Position in Generic Letter 83-28, Item 2.2 Part 2 'Vendor Interface for Safety-Related Components' (Generic Letter No. 90-03)," March 20, 1990.

2. INPO 84-010, "Vendor Equipment Technical Information Program," Nuclear Utility Task Action Committee, March 1984.

(MPA No. B-078) TITLE: Items 3.1.1 and 3.1.2 - Post-Maintenance Testing (Reactor Trip System Components)

Items 3.1.1 and 3.1.2 concern post-maintenance testing procedures and vendor recommendations for reactor trip system components. Licensees and applicants

are to review their test and maintenance procedures and Technical Specifications to assure that they require post-maintenance operability testing of safety-related components in the reactor trip system and that such testing demonstrates that the equipment is capable of performing its safety functions prior to returning it to service. Licensees and applicants are also to review applicable vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or in the Technical Specifications, where required. The results of these reviews are to be submitted for staff evaluation.

(MPA No. B-079) TITLE: Item 3.1.3 - Post-Maintenance Testing - Changes to Test Requirements (Reactor Trip System Components)

Item 3.1.3 requests identification of any applicable post-maintenance test requirements in existing Technical Specifications for reactor trip system components which can be demonstrated to degrade rather than enhance safety. Licensees and applicants are to perform the required reviews and notify the staff of their findings. Appropriate changes to these test requirements, with supporting justification, are to be submitted for staff approval.

(MPA No. B-087) TITLE: Items 3.2.1 and 3.2.2 - Post-Maintenance Testing (All Other Safety-Related Components)

Items 3.2.1 and 3.2.2 concern post-maintenance testing procedures and vendor recommendations for all safety-related components other than the reactor trip system components. Licensees and applicants are to review their test and maintenance procedures and Technical Specifications to assure that they require post-maintenance operability testing of all safety-related components (non-reactor trip system components) and that such testing demonstrates that the equipment is capable of performing its safety functions prior to returning it to service. Licensees and applicants are also to review applicable vendor and engineering recommendations to ensure that any appropriate guidance is included in the test and maintenance procedures or in the Technical Specifications, where required. The results of these reviews are to be submitted for staff evaluation.

(MPA No. B-088) TITLE: Item 3.2.3 - Post-Maintenance Testing - Changes to Test Requirements (All Other Safety-Related Components)

Item 3.2.3 requests identification of any applicable post-maintenance test requirements in existing Technical Specifications for safety-related components which can be demonstrated to degrade rather than enhance safety. Licensees and

applicants are to perform the required reviews and notify the staff of their findings. Appropriate changes to these test requirements, with supporting justification, are to be submitted for staff approval.

(MPA No. B-080) TITLE: Item 4.1 - Reactor Trip System Reliability (Vendor-Related Modifications)

The resolution of this item, applicable to all plants, requests that each licensee review all vendor-recommended reactor trip breaker modifications to verify that either: (1) each modification has been implemented, or (2) a written evaluation of the technical reasons for not implementing a modification exists. Submittals were to be made by all licensees/applicants. For those plants that were licensed at the time, the submittals were to be reviewed by the cognizant regions and Safety Evaluations were issued by NRR. For plants licensed since 1983, Item 4.1 was to be included as part of the licensing review and the results reported in the licensing SER or in one of the supplements.

(MPA No. B-081) TITLE: Items 4.2.1 and 4.2.2 - Reactor Trip System Reliability - Maintenance and Testing (Preventative Maintenance and Surveillance Program for Reactor Trip Breakers)

Item 4.2.1 addresses development of a planned program of periodic maintenance, including lubrication, housekeeping and other items recommended by the equipment suppliers. Item 4.2.2 addresses development and implementation of a program for trending of parameters which affect breaker operation and are measured during testing in order to predict performance degradation. All PWR licensees and applicants were to provide descriptions of their programs for staff review.

(MPA No. B-082) TITLE: Item 4.3 - Reactor Trip System Reliability-Design Modifications (Automatic Actuation of Shunt Trip Attachment for Westinghouse and B&W Plants)

This portion of Item 4.3 requests all licensees and applicants with Westinghouse and B&W plants to modify their reactor trip systems to provide automatic actuation of the breaker shunt trip attachments. The staff was to review the submittals and issue SERs for all affected plants.

(MPA No. B-090) TITLE: Item 4.3 - Reactor Trip System Reliability - Technical Specification Changes (Automatic Actuation of Shunt Trip Attachment for Westinghouse and B&W Plants)

This portion of Item 4.3 requests submittal of Technical Specifications changes addressing the implementation of automatic actuation of the breaker

shunt trip attachments on all Westinghouse and B&W plants (see previous discussion on MPA No. B-082). The staff developed model Technical Specifications and transmitted them to affected licensees and applicants in Generic Letter 85-09. The staff was to review the submittals and issue license amendments and/or SERs for all affected plants.

Additional Reference:

1. NRC Letter to All Westinghouse Pressurized Water Reactor Licensees and Applicants, "Technical Specifications for Generic Letter 83-28, Item 4.3 (Generic Letter 85-09)," dated May 23, 1985.

(MPA No. B-091) TITLE: Item 4.4 - Reactor Trip System Reliability (Improvements in Maintenance and Test Procedures for B&W Plants)

Item 4.4 requests B&W reactor licensees and applicants to apply safety-related maintenance and test procedures to the diverse reactor trip feature provided by interrupting power to control rods through the silicon controlled rectifiers (SCRs). Specifically, licensees and applicants are requested to submit for staff review: (1) confirmation that procedures which comply with all requirements of safety-related procedures are being used to maintain and test the SCRs, (2) a brief description of the procedures used to conduct periodic surveillance, testing and maintenance of the SCR diverse reactor trip feature; such tests should verify that the SCRs under test have degated and opened the power supply circuit to the control rod holding coils, and (3) Technical Specifications changes which include requirements for safety-related surveillance and tests of the SCRs to be performed at intervals commensurate with existing test intervals for other safety-related portions of the reactor trip system or alternatively, show that these requirements are in the existing Technical Specifications.

(MPA No. B-092) TITLE: Item 4.5.1 - Reactor Trip System Reliability-Diverse Trip Features (System Functional Testing)

Item 4.5.1 requests that licensees perform on-line functional testing of the reactor trip system, including independent testing of the diverse trip features. The diverse trip features to be tested include the breaker undervoltage and shunt trip features on Westinghouse, B&W and CE plants; the circuitry used for power interruption with the silicon controlled rectifiers on B&W plants; and the scram pilot valves and backup scram valves (including all initiating circuitry) on GE plants.

Licensees were requested to confirm that the required on-line functional surveillance testing is being performed for the diverse trip features of the plant.

Some licensees do not test backup scram valves on-line, because such testing would result in a reactor scram. In such cases the NRC allows scram valves to be tested during each refueling outage to avoid unnecessary reactor scrams and challenges to the reactor protection system. Conformance with this item is verified by follow-up inspections.

(MPA No. B-093) TITLE: Items 4.5.2 & 4.5.3 - Reactor Trip System Reliability - Test Alternatives and Intervals (System Functional Testing)

Item 4.5.2 requests licensees and applicants to certify whether their plants are designed to permit on-line functional testing of the reactor trip system (RTS). For plants not designed to permit such testing, licensees are requested to commit to design modifications which would permit such testing and provide an implementation schedule, or to provide justification for not implementing on-line testing capability. The staff will consider alternatives to on-line testing where special circumstances exist and where the objective of high reliability can be met by other means.

Item 4.5.3 requests licensees and applicants to confirm that on-line functional testing of the RTS is being performed and that existing test intervals required by their Technical Specifications are adequate for achieving high RTS reliability. All four vendors submitted topical reports which presented analyses demonstrating that current test intervals provide high reliability. Based on staff review of the Owner's Group topical reports, the contractors' independent analyses, and the generic safety evaluation findings in NUREG-0460, the staff concluded that the existing intervals, as recommended in the topical reports, for on-line functional testing are consistent with achieving high RTS availability at all operating reactors. Licensees and applicants are to submit a description of how they are implementing the provisions of their Owner's Group topical report.

Additional References:

1. Topical Report WCAP-10271, "Evaluation of the Surveillance Frequencies and Out of Service Times for the Reactor Protection Systems," 1985.

2. Topical Report WCAP-10271, Supplement 1.

3. NECD-30844, "BWR Owner's Group Response to NRC Generic Letter 83-28, Item 4.5.3," January 1985.

4. NECD-30851P, "Technical Specification Improvement Analyses for BWR Reactor Protection System," May 1985.

5. CE NPSD-277, "Reactor Protection System Test Interval Evaluation, Task 486," December 1984.

6. BAW-10167, "Justification for Increasing the Reactor Trip System On-Line Test Interval," May 1986.

7. BAW-10167, Supplement 1, February 1988.

GSI No. 86 (MPA No. B-084) TITLE: Long Range Plan for Dealing With Stress Corrosion Cracking in BWR Piping

This issue arose from inspections conducted at several boiling water reactors (BWRs) which revealed intergranular stress corrosion cracking (IGSCC) in large-diameter recirculation and residual heat removal piping. These inspections were conducted pursuant to IE Bulletins 82-03, 82-03 Revision 1, and 83-02 and the NRC August 26, 1983 Orders. The Commission concluded that the results of these inspections mandated an ongoing program for similar reinspections at all operating BWRs.

Generic Letter 84-11 requested all BWR licensees and applicants to submit, for staff review, their plans and surveillance measures relative to the staff positions set forth in the Generic Letter and to commit to develop and implement an acceptable program to detect potential IGSCC.

Inspections conducted pursuant to GL 84-11 disclosed a significant number of cracks in BWR piping. The staff concluded that augmented inspections and licensee actions beyond those in GL 84-11 were warranted. Generic Letter 88-01 was subsequently issued describing the staff's revised positions on what were acceptable actions that licensees/applicants should take to minimize the potential for IGSCC. The staff positions in GL 88-01 superseded those in GL 84-11 and are beyond the scope of this GSI.

References:

1. IE Bulletin 82-03 "Stress Corrosion Cracking in Thick-Wall, Large Diameter, Stainless Steel, Recirculation System Piping at BWR Plants," U. S. Nuclear Regulatory Commission, October 14, 1982.

2. IE Bulletin 82-03, Revision 1, October 28, 1982.

3. IE Bulletin 83-02, "Stress Corrosion Cracking in Large Diameter Stainless Steel Recirculation Systems Piping at BWR Plants," U.S. Nuclear Regulatory Commission, March 4, 1983.

4. NRC Letter to All Licensees of Operating Reactors, Applicants for Operating License, and Holders of Construction Permits for Boiling Water Reactors, "Inspections of BWR Stainless Steel Piping," (Generic Letter 84-11), April 19, 1984.

5. NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," U.S. Nuclear Regulatory Commission, July 1977; Rev. 1, July 1980; Rev. 2, January 1988.

6. NRC Letter to All Licensees of Operating Boiling Water Reactors (BWRs), and Holders of Construction Permits, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping (Generic Letter 88-01)," January 25, 1988.

GSI No. 93 (MPA No. B-098) TITLE: Steam Binding of Auxiliary Feedwater Pumps

The issue concerns the potential disabling of auxiliary feedwater pumps by steam binding caused by back-leakage of main feedwater past the isolation check valves. IE Bulletin 85-01, issued October 29, 1985, requested that certain licensees implement procedures for monitoring the auxiliary feedwater piping temperatures for indications of possible back-leakage and for restoring the pumps to operable status if steam binding were to occur.

Generic Letter 88-03, issued February 17, 1988, stated that the plants that received Bulletin 85-01 should continue following the Bulletin's recommendations, and requested that these recommendations be followed on all PWR's.

References:

1. IE Bulletin No. 85-01, "Steam Binding of Auxiliary Feedwater Pumps," U.S. Nuclear Regulatory Commission, October 29, 1985.

2. NRC Letter to All Licensees, Applicants for Operating Licenses, and Holders of Constructions Permits for Pressurized Water Reactors, "Resolution of Generic Safety Issue 93, 'Steam Binding of Auxiliary Feedwater Pumps' (Generic Letter 88-03)," February 17, 1988.

GSI No. 99 (MPA No. L-817) TITLE: RCS/RHR Suction Line Valve Interlock on PWRs

This issue concerns the inadvertent closing of RHR suction valves when the RHR system is in use.

Interlocks are provided on these valves to ensure that a double barrier (i.e., two closed valves) is maintained between the RCS and RHR systems when the plant is at normal operating conditions. However, the loss of one instrument bus or disturbance of one logic channel will result in the automatic closure of one of the RHR suction line isolation valves. Such closure gives rise to the potential for RHR pump damage and loss of decay heat removal capability if the RHR pump is not interlocked with the RHR suction valves.

The scope of this issue was broadened in June 1986 to include the less frequent but higher risk mode of failure associated with mid-loop operation. Generic Letter 87-12 addressed this concern.

Generic Letter 88-17 superseded GL 87-12 and requested responses regarding licensee plans with respect to operation on shutdown cooling. This letter requested expeditious licensee actions in the areas of: (1) training of operators before entering a reduced inventory condition, (2) implementation of procedures and administrative controls related to decay heat removal, (3) temperature and level indications, and (4) alternate means of adding water to the RCS. Further,

GL 88-17 identified a number of programmed enhancements to be developed in the following six areas: (1) instrumentation, (2) procedures, (3) equipment, (4) analyses, (5) Technical Specifications, and (6) RCS perturbations.

References:

1. NRC Letter to All Licensees of Operating PWRs and Holders of Construction Permits for PWRs, "Loss of Residual Heat Removal (RHR) While the Reactor Coolant System (RCS) is Partially Filled (Generic Letter 87-12)," July 9, 1987.

2. NRC Letter to All Holders of Operating Licenses or Construction Permits for Pressurized Water Reactors (PWRs), "Loss of Decay Heat Removal (Generic Letter No. 88-17), 10 CFR 50.54(f)," October 17, 1988.

3. NUREG/CR-5015, "Improved Reliability of Residual Heat Removal Capability in PWRs as Related to Resolution of Generic Issue 99," U.S. Nuclear Regulatory Commission, May 1988.

GSI No. 124 TITLE: Auxiliary Feedwater System Reliability

This issue was initially established after implementation of upgrades to the auxiliary feedwater (AFW) systems in all PWR plants, under TMI Action Plan Clarification, NUREG-0737, Items II.E.1.1 and I.E.1.2, in order to determine if further improvements in AFW system reliability were necessary. NUREG-0737, Items II.E.1.1 and II.E.1.2 addressed implementation of recommendations for improving AFW system reliability identified in NUREG-0611 and -0635.

Based on evaluation of AFW system reliability studies for various plants, the staff subsequently determined that three-pump AFW systems demonstrated significantly greater reliability than did two-pump systems and, therefore, limited this issue to those two-pump plants for which the licensee had not committed to add a third means of delivering water to the steam generators for post-transient/accident decay heat removal. The affected plants are ANO-1 & 2, Rancho Seco, Prairie Island 1 & 2, Crystal River-3 and Ft. Calhoun.

The staff performed plant-specific reviews of the reliability of the AFW systems in the above plants, including assessments of the system design, operating experience, and emergency procedures. From these reviews, the staff determined whether additional means of secondary decay heat removal capability was necessary. No further hardware modification was determined to be required for ANO-1 and Prairie Island 1 & 2 on the basis of the startup feedwater pump and AFW system sharing capability, respectively. The licensees for Rancho Seco, Crystal River-3 and Ft. Calhoun committed to install additional means of secondary decay heat removal, thereby resolving the issue. The staff issued a plant-specific backfit analysis for ANO-2 requiring the addition of a third train of secondary decay heat removal. Implementation of the modifications to these plants is proceeding in accordance with schedules agreed to by the staff.

References:

1. NUREG-0611, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants," U.S. Nuclear Regulatory Commission, January 1980.

2. NUREG-0635, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Combustion Engineering Designed Operating Plants," U.S. Nuclear Regulatory Commission, January 1980.

3. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980.

GSI No. A-13 (MPA No. B-017) TITLE: Snubber Operability Assurance-Hydraulic Snubbers

This issue concerns operability of hydraulic snubbers which is required to assure that the structural integrity of the reactor coolant system is maintained during and following a seismic or other event initiating dynamic loads. Operating experience in the 1970's indicated the need for changes, clarifications and improvements in snubber Technical Specifications. These changes provided for: (1) precluding use of an arbitrary snubber capacity as a limit for inservice test requirements, (2) elimination of the requirement that seal material be approved by NRC, (3) implementation of a monitoring program to assure snubber reliability, (4) development and implementation of clearly defined inservice test requirements, and (5) permissible in-place inservice testing.

By letter dated November 20, 1980, the NRC requested that all power reactor licensees (except Systematic Evaluation Program (SEP) licensees) incorporate the above changes in plant-specific Technical Specifications. A similar request was sent to SEP licensees on March 23, 1981. Also, revisions to the Standard Technical Specifications (W, GE, CE and BW) incorporated the appropriate Technical Specifications to address these changes for NTOLs.

References

1. NRC Letter to All Power Reactor Licensees (Except SEP Licensees), "Technical Specification Revisions for Snubber Surveillance," November 20, 1980.

2. NRC Letter to all SEP Power Reactor Licensees, (Except SEP Licensees), "Technical Specification Revisions for Snubber Surveillance," March 23, 1981.

GSI No. A-13 (MPA No. B-022) TITLE: Snubber Operability Assurance-Mechanical Snubbers

This aspect of the issue addresses mechanical snubbers. In the mid 1970's, several deficiencies were noted in the Technical Specifications for assuring snubber reliability. Also, mechanical snubbers were not included in the Technical Specifications surveillance requirements. Many licensees used mechanical snubbers as original equipment and others requested to replace their hydraulic snubbers with mechanical ones to simplify or avoid inservice surveillance. The most likely failure for an unsurveilled mechanical snubber is permanent lock-up which can be harmful to plant systems during normal operations and during seismic events initiating dynamic loads. Therefore, changes were needed which would: (1) include mechanical snubbers in the surveillance program, (2) preclude use of an arbitrary snubber capacity as a limit for inservice test requirements, (3) implement a monitoring program to assure snubber reliability, (4) develop and implement clearly defined test requirements, and (5) permit in-place inservice testing.

By letter dated November 20, 1980, the NRC requested that all power reactor licensees (except Systematic Evaluation Program (SEP) licensees) incorporate the above changes in plant-specific Technical Specifications. A similar request was sent to SEP licensees on March 23, 1981. Also, revisions to the Standard Technical Specifications (W, GE, CE and BW) incorporated the appropriate Technical Specifications to address these changes for NTOLs.

References:

1. NRC Letter to All Power Reactor Licensees (Except SEP Licensees), "Technical Specification Revisions for Snubber Surveillance," November 20, 1980.

2. NRC Letter to All SEP Power Reactor Licensees, "Technical Specification Revisions for Snubber Surveillance," March 23, 1981.

GSI No. A-16 (MPA No. D-012) TITLE: Steam Effects on BWR Core Spray Distribution

This issue arose from tests which showed that the presence of steam and/or increased pressure in and above the upper core region of BWRs could adversely affect the distribution of flow from certain types of core spray nozzles following a LOCA. The distribution that had been assumed in BWR LOCA analyses was based on tests of core spray nozzles conducted by GE in an air (non-steam) environment.

In response to staff concerns regarding the core spray performance, GE took the lead for resolving the issue generically. This GSI was established for staff review of generic actions. However, because of design differences among the various BWR product lines, resolution of this issue has taken different forms for different classes of BWR plants. Each of the different resolution paths is summarized below.

1. BWR/1 (Big Rock Point) - The licensee performed a test with the installed core spray system which demonstrated the adequacy of the spray flow distribution. The staff found the test results acceptable and concurred in the licensee's resolution of this issue in 1979.

2. BWR/2 (Oyster Creek and Nine Mile Point 1) - Core spray is less important in plants with jet pumps because these plants are designed to reflood to 2/3 core height during a LOCA and cooling over that 2/3 height is effective. Non-jet pump plants of the BWR/2 design do not reflood for large breaks below the core, and must rely on adequate spray flow to each assembly and steam cooling to avoid fuel melt. Because of this concern, letters were issued to the two BWR/2 licensees requesting that they justify the spray cooling (heat transfer) coefficients assumed in their ECCS analyses. Multi-plant Action (MPA) D-012 was established for the review of the core spray issue on these two BWR/2 plants, the only plants which were affected by GSI A-16. In both cases, the licensees, with assistance from GE, were able to show to the staff's satisfaction that even when including the effects of a steam environment on core spray distribution, the degraded distribution of core spray along with steam cooling was adequate to ensure that clad temperature limits specified in 10 CFR 50.46 would not be exceeded.

3. BWR/3/4/5 - In reviewing the core spray distribution issue for these jet pump plant class designs, the staff found that as long as the reflood water level could be maintained with injection from the core spray system, the distribution of core spray over the top of the core was not a significant factor in achieving adequate core cooling. For this reason the core spray distribution issue was resolved generically for these plants and no actions were requested of these licensees.

4. BWR/6 - To resolve the core spray distribution issue for the BWR/6 design, GE performed full-scale tests of the BWR/6 core spray sparger in a steam environment. The staff inspected the GE test facility, reviewed the test results and concluded that the BWR/6 core spray design was adequate. This resolved the core spray distribution issue for the BWR/6 design.

References:

1. Letter to I.R. Finfrock, Jr., JCP&L Co. from George Lear, NRC; December 10, 1976.

2. Letter to NMPC from NRC dated December 10, 1976.

GSI No. A-35 (MPA No. B-023) TITLE: Adequacy of Offsite Power Systems

This issue arose from a July 1976 degraded grid voltage condition which occurred at Millstone 2 and which resulted in blown fuses in certain engineered safety feature equipment. As a result, the staff determined that a potential

existed for supplying both safety and non-safety equipment with voltages outside the design range, which could render the equipment inoperable.

Letters were sent to licensees in June 1977 which requested installation of degraded voltage relays designed to separate the safety buses from offsite power whenever the degraded voltage condition existed for more than about 10 seconds. Licensees were also requested to propose Technical Specifications with LCOs and surveillance requirements for these relays and associated instrumentation. Some licensees chose instead to institute procedures for manual actions in the event of these degraded voltage conditions. The Regions reviewed these procedures and eventually found them to be acceptable.

Some licensees resolved this issue in conjunction with MPA B-048, "Adequacy of Station Distribution Voltage," which was initiated by the letter to all power reactor licensees (except Humboldt Bay) on August 8, 1979. MPA B-048 requested licensees to reanalyze their plants to ensure that safety-related equipment was not subjected to voltages outside design limitations when the grid voltage was at its maximum and minimum levels. After performing these analyses, licensees were then to perform a test to measure station voltages at various places in the plant to verify the accuracy of the calculations. As a result of this review, many licensees made tap changes to transformers to optimize station distribution voltages. These tap changes often affected MPA-B023 calculations and caused changes to the undervoltage relay setpoints.

The changes imposed by resolution of this issue were incorporated into licensing reviews after 1977 through Branch Technical Position PSB-1 and, subsequently, a 1981 revision to SRP 8.3.1, Appendix A.

References:

1. NRC Letter to Northeast Nuclear Energy Company, "Millstone Nuclear Power Station Unit Nos. 1 and 2," June 2, 1977.

2. NRC Letter to All Power Reactor Licensees (Except Humboldt Bay), "Adequacy of Station Electric Distribution Systems Voltages," August 8, 1979.

3. Branch Technical Position PSB-1, "Adequacy of Station Electric Distribution Voltages," July 1981.

GSI No. B-10 TITLE: Behavior of BWR Mark III Containments

This GSI involved completion of the staff evaluation of the Mark III containment loads and documentation of the method used to validate the analytical models and assumptions needed to predict the containment pressure loads in the event of a LOCA. The BWR Mark III containment design differed from the previously-reviewed Mark I and Mark II designs. As a result, staff

acceptance criteria were required for the various pool dynamic loads associated with this new design.

The Mark III suppression pool dynamic loads were reviewed by the staff at the CP stage for the Grand Gulf Nuclear Station and at the preliminary design analysis (PDA) stage for GESSAR-238NI. The information available was deemed sufficient to adequately define the pool dynamic loads for those Mark III nuclear plants under review for CPs. Since the issuance of the GESSAR-238NI SER in Decmber 1975, GE has conducted further tests and analyses to confirm and refine the original load definitions. The GESSAR-II FDA application provides the finalized pool dynamic load definition for Mark III containments and associated piping and is the basic document used for review by the staff.

The staff has published the results of its generic review in NUREG-0978. Revision 6 to the SRP Section 6.2.1.1.C states that the acceptability of pool dynamic loads for plants with Mark III containments is based on conformance with the NRC acceptance criteria identified in Appendix C of NUREG-0978. The plant-specific design for all Mark III plants was reviewed at the time of licensing, using this NUREG as the staff's acceptance criteria and the results were to be documented in the SER of each Mark III plant.

References:

1. NUREG-0978, "Mark III LOCA-Related Hydrodynamic Load Definition," U.S. Nuclear Regulatory Commission, February 1984.

2. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C and D Tasks)," U.S. Nuclear Regulatory Commission, June 1978.

GSI No. B-36 TITLE: Develop Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and for Normal Ventilation Systems

This issue is concerned with the implementation of criteria for the design, testing and maintenance of air filtration and adsorption equipment. The criteria were published in Revision 2 of Regulatory Guide 1.52 and in Revision 1 of Regulatory Guide 1.140.

The major changes in the criteria for this type of equipment, applicable to all plants with operating license applications after April 1, 1980, from previous requirements were in the provision of redundant protection against particulate release resulting from a HEPA filter failure, the requirement that equipment be designed for the expected range of temperature and other environmental conditions such as radiation, the use of both heating and cooling for humidity control, the use of type-tested fan motors, automatic initiation, testing of carbon and carbon performance requirements, provision of adequate drains, and access requirements and physical external clearances for removal and replacement of internals. These revised criteria also superseded those in ORNL-NSIC-65.

References:

1. Regulatory Guide 1.52, Rev. 2, "Design, Testing and Maintenance Criteria for Post-Accident Engineered Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," March 1978.

2. Regulatory Guide 1.140, Rev. 1, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," October 1979.

GSI No. B-63 (MPA No. B-045) TITLE: Isolation of Low Pressure Systems Connected to the Reactor Coolant System Pressure Boundary

This issue resulted from staff concerns regarding the potential failure of valves comprising the pressure isolation barrier between the reactor coolant system (RCS) and interfacing low-pressure systems. Such a failure could result in overpressurization and attendant rupture of low-pressure piping and/or components, with a loss of coolant outside containment. The Reactor Safety Study (WASH-1400) identified the intersystem loss-of-coolant accident (ISLOCA) in PWRs as a significant contributor to risk from core melt. The study focused on two specific pressure isolation configurations consisting of two in-series check valves, with or without an open motor-operated valve in series. This accident scenario was designated as Event V.

Concerns regarding Event V, as well as the staff's position that valve closure integrity could be improved by testing, led to the issuance of a Generic Letter entitled "LWR Primary Coolant System Pressure Isolation Valves," dated February 23, 1980, which requested a response from all licensees specifying whether their facilities contained the Event V configurations.

For the 34 facilities (32 PWRs, 2 BWRs) responding affirmatively, orders were issued on April 20, 1981 imposing certain corrective actions, including implementation of periodic testing of the identified Event V pressure isolation valves (PIVs) and Technical Specifications addressing surveillance and limiting conditions of operation for these PIVs.

References:

1. WASH-1400 (NUREG-75/014), "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, October 1975.

2. NRC Letter to All LWR Licensees, "LWR Primary Coolant System Pressure Isolation Valves," February 23, 1980.

LIST OF RECENTLY ISSUED GENERIC LETTERS

Generic Letter No.	Subject	Date of Issuance	Issued To
89-13 SUPP. 1	SERVICE WATER SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT	04/04/90	ALL HOLDERS OF OLs OR CPs FOR NUCLEAR POWER PLANTS
88-20, SUPP. 2	ACCIDENT MANAGEMENT STRATEGIES FOR CONSIDERATION IN THE INDIVIDUAL PLANT EXAM PROCESS	04/04/90	ALL HOLDERS OF OLs AND CPs FOR NUCLEAR POWER REACTOR FACILITIES
90-03	RELAXATION OF STAFF POSITION IN GL 83-28, ITEM 2.2, PART 2 "VENDOR INTERFACE FOR SAFETY-RELATED COMPONENTS"	03/20/90	ALL POWER REACTOR LICENSEES AND APPLICANTS
90-02	ALTERNATIVE REQUIREMENTS FOR FUEL ASSEMBLIES IN THE DESIGN FEATURES SECTION OF TECHNICAL SPECIFICATIONS	02/01/90	ALL LWR LICENSEES AND APPLICANTS
90-01	REQUEST FOR VOLUNTARY PARTICIPATION IN NRC REGULATORY IMPAC SURVEY	01/18/90	ALL LICENSEES OF OPERATING REACTORS & CONSTRUCTION PERMITS FOR LWR NUCLEAR POWER PLANTS
89-23	NRC STAFF RESPONSES TO QUESTIONS PERTAINING TO IMPLEMENTATION OF 10 CFR PART 26 - GENERIC LETTER 89-23	10/23/89	ALL HOLDERS OF OPERATING LICENSEES AND CONSTRUCTION PERMITS FOR NUCLEAR POWER PLANTS
89-22	POTENTIAL FOR INCREASED ROOF LOADS AND PLANT AREA FLOOD RUNOFF DEPTH AT LICENSED NUCLEAR POWER PLANTS DUE TO RECENT CHANGE IN PROBABLE MAXIMUM PRECIPITATION CRITERIA DEVELOPED BY THE NATIONAL WEATHER SERVICE	10/19/89	ALL LICENSEES OF OPERATING REACTORS AND HOLDERS OF CONSTRUCTION PERMITS (EXCEPT BYRON BRAIDWOOD, VOGTLE, SOUTH TEXAS, AND RIVER BEND)

suggestions for reducing this burden, to the Information and Records Management Branch, Division of Information Support Services, Office of Information Resources Management (MNBB-7714), U.S. Nuclear Regulatory Commission, Washington, D.C. 20555; and to the Paperwork Reduction Project (3150-0011), Office of Management and Budget, Washington, D.C. 20503.

Please address your response to this generic letter to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, pursuant to 10 CFR Section 50.4 of the NRC's regulations.

Sincerely,

Original signed by
James G. Partlow
James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. GSI Table
2. GSI Summaries
3. List of Most Recently Issued NRC Generic Letters

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