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

October 19, 1989

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

SUBJECT: REQUEST FOR INFORMATION CONCERNING STATUS OF IMPLEMENTATION OF UNRESOLVED SAFETY ISSUE (USI) REQUIREMENTS (GENERIC LETTER 89-21)

As part of our continuing effort to validate staff understanding regarding implementation of significant regulatory issues, the staff is conducting a comprehensive review of the implementation status of unresolved safety issues (USIs). An important aspect of this effort is to ensure that the licensee and NRC agree on the status of USI resolution implementation at each facility. The purpose of this letter is to request your review and reporting of the status of implementation of USIs for which a final technical resolution has been achieved and which are applicable to your facility.

To assist you in this effort, I have enclosed a table of USIs for which a final technical resolution has been achieved (Enclosure 1). This table indicates other information, such as multiplant action (MPA) number, generic letter number, applicability, and reference NUREG number. For your facility, determination of requirements for a particular USI may necessitate review of applicable generic letters, NUREG documents, or plant-specific correspondence. For your information, a summary of the resolution of each USI is provided in Enclosure 2.

As in the case of our earlier correspondence related to the status of implementation of TMI Task Action Plan items, implementation should be considered complete when activities have been performed necessary to satisfy the requirements (or assumptions) made in the staff's technical resolution of the particular USI. If you have not fully completed an item, we ask you to mark up the enclosure to reflect your projected implementation date. You should add a short note identifying remaining work (e.g., hardware, procedures, training, technical specifications). More explicit instructions are provided as part of Enclosure 1.

Your NRC Project Manager is developing data sheets that identify significant plant-specific correspondence between each licensee and the staff relating to a particular USI. Once we have researched agency files we will provide this information to your staff. This will ensure we both have a clear record of major actions regarding the USI. The Project Manager can provide additional clarification which may be of assistance to you and will work with your staff to identify plant-specific references.

We request that this information be provided within 30 days of receipt of this letter. The information we are requesting will be utilized to validate and update our existing databases so that we will have an accurate and complete understanding of the status of USI implementation at each nuclear power plant.

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This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires December 31, 1989. The estimated average burden hours is 80 person hours per plant, including searching data sources, gathering and analyzing the data, and preparing the required letter. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Records and Reports Management Branch, Division of Information Support Services, Office of Information Resources Management, 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.

Sincerely,

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

Enclosures:
1. USI Table
2. USI Issues Summary
3. List of Most Recently Issued NRC Generic Letters

ENCLOSURE 1

GUIDE FOR UPDATING USI STATUS

(1) Enclosure 1 lists all unresolved safety issues (USIs) for which a final technical resolution has been achieved. Please review the entire listing for each licensed reactor unit. Where an item is not applicable for your facility, mark "NA" in the status column.

(2) Where an item is applicable to your facility, but no changes were necessary, mark "NC" in the status column.

(3) Where an item is applicable to your facility and changes are complete, mark "C" in the status column and indicate month and year implementation was complete, including reference to any supporting documentation.

(4) Where an item is applicable to your facility and is not fully implemented, provide your projected implementation date (month and year) and a short note identifying the outstanding item (e.g., hardware, procedures, training, Technical Specifications). Mark "I" for incomplete.

(5) Where a USI resolution was only recently issued, such as A-40 and A-47, and you are evaluating your response, identify expected response date and indicate "E" in the status column.

ENCLOSURE 1

UNRESOLVED SAFETY ISSUES FOR WHICH A FINAL TECHNICAL RESOLUTION HAS BEEN ACHIEVED

USI/MPA NUMBER	TITLE	REF. DOCUMENT	APPLICABILITY	STATUS/DATE*	REMARKS
A-1	Water Hammer	SECY 84-119 NUREG-0927, Rev. 1 NUREG-0993, Rev. 1 NUREG-0737 Item I.A.2.3 SRP revisions	All		
A-2/ MPA D-10	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	NUREG-0609 GL 84-04, GDC-4	PWR		
A-3	Westinghouse Steam Generator Tube Integrity	NUREG-0844 SECY 86-97 SECY 88-272 GL 85-02 (No requirements)	W-PWR		
A-4	CE Steam Generator Tube Integrity	NUREG-0844, SECY 86-97 SECY 88-272 GL 85-02 (No requirements)	CE-PWR		
A-5	B&W Steam Generator Tube Integrity	NUREG-0844, SECY 86-97 SECY 88-272 GL 85-02 (No Requirements)	B&W-PWR		
E A-6	Mark I Containment Short-Term Program	NUREG-0408	Mark I-BWR		

* C - COMPLETE
NC - NO CHANGES NECESSARY
NA - NOT APPLICABLE
I - INCOMPLETE
E - EVALUATING ACTIONS REQUIRED

USI/MPA NUMBER	TITLE	REF. DOCUMENT	APPLICABILITY	STATUS/DATE*	REMARKS
A-7/ D-01	Mark I Long-Term Program	NUREG-0661 NUREG-0661 Suppl. 1 GL 79-57	Mark I-BWR		
A-8	Mark II Containment Pool Dynamic Loads	NUREG-0808 NUREG-0487, Suppl. 1/2 NUREG-0802 SRP 6.2.1.1C GDC 16	Mark II-BWR		
A-9	Anticipated Transients Without Scram	NUREG-0460, Vol. 4 10 CFR 50.62	All		
A-10/ MPA B-25	BWR Feedwater Nozzle Cracking	NUREG-0619 Letter from DG Eisenhut dated 11/13/80 GL 81-11	BWR		
A-11	Reactor Vessel Material Toughness	NUREG-0744, Rev. 1 10 CFR 50.60/ 82-26	All		
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	NUREG-0577, Rev. 1 SRP Revision 5.3.4	PWR		
A-17	Systems Interactions	Ltr: DeYoung to licensees - 9/72 NUREG-1174, NUREG-1229, NUREG/CR-3922, NUREG/CR-4261, NUREG/CR-4470, GL 89-18 (No requirements)	All		
A-24/ MPA B-60	Qualification of Class 1E Safety-Related Equipment	NUREG-0588, Rev. 1 SRP 3.11 10 CFR 50.49 GL 82-09, GL 84-24 GL 85-15	All		

USI/MPA NUMBER	TITLE	REF. DOCUMENT	APPLICABILITY	STATUS/DATE*	REMARKS
A-26/ MPA B-04	Reactor Vessel Pressure Transient Protection	DOR Letters to Licensees 8/76 NUREG-0224 NUREG-0371 SRP 5.2 GL 88-11	PWR		
A-31	Residual Heat Removal Shutdown Requirements	NUREG-0606 RG 1.113, RG 1.139 SRP 5.4.7	All OLs After 01/79.		
A-36/ C-10, C-15	Control of Heavy Loads Near Spent Fuel	NUREG-0612 SRP 9.1.5 GL 81-07, GL 83-42, GL 85-11 Letter from DG Eisenhut dated 12/22/80	All		
A-39	Determination of SRV Pool Dynamic Loads and Pressure Transients	NUREG-0802 NUREGs-0763,0783,0802 NUREG-0661 SRP 6.2.1.1.C	BWR		
A-40	Seismic Design Criteria	SRP Revisions, NUREG/ CR-4776, NUREG/CR-0054, NUREG/CR-3480, NUREG/ CR-1582, NUREG/CR-1161, NUREG-1233, NUREG-4776 NUREG/CR-3805 NUREG/CR-5347 NUREG/CR-3509	All		
A-42/ MPA B-05	Pipe Cracks in Boiling Water Reactors	NUREG-0313, Rev. 1 NUREG-0313, Rev. 2 GL 81-03, GL 88-01	BWR		

USI/MPA NUMBER	TITLE	REF. DOCUMENT	APPLICABILITY	STATUS/DATE*	REMARKS
A-43	Containment Emergency Sump Performance	NUREG-0510, NUREG-0869, Rev. 1 NUREG-0897, R.G.1.82 (Rev. 0), SRP 6.2.2 GL 85-22 No Requirements	All		
A-44	Station Blackout	RG 1.155 NUREG-1032 NUREG-1109 10 CFR 50.63	All		
A-45	Shutdown Decay Heat Removal Requirements	SECY 88-260 NUREG-1289 NUREG/CR-5230 SECY 88-260 (No requirements)	All		
A-46	Seismic Qualification of Equipment in Operating Plants	NUREG-1030 NUREG-1211/ GL 87-02, GL 87-03	All		
A-47	Safety Implication of Control Systems	NUREG-1217, NUREG-1218 GL 89-19	All		
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	10 CFR 50.44 SECY 89-122	All, except PWRs with large dry containments		
A-49	Pressurized Thermal Shock	RGs 1.154, 1.99 SECY 82-465 SECY 83-288 SECY 81-687 10 CFR 50.61/ GL 88-11	PWR		

on July 23, 1986. In addition, it has also been satisfactorily demonstrated in the course of the A-2 effort that there is a very low likelihood of simultaneous pipe loading with both LOCA and safety shutdown earthquake (SSE) loads. Therefore, the last revision of GDC-4 represented the final technical action of NRC regarding the issue of asymetric blowdown loads issue in PWRs primary coolant main loop piping.

3. USI NO. A-3,4,5 TITLE: Steam Generator Tube Integrity

USIs A-3, 4, and 5, were resolved in September 1988 with the publication of NUREG-0844 "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." USIs A-3, A-4, and A-5 did not result in new generic requirements for industry in view of the small potential for reducing risk.

Steam generator tube integrity was designated an unresolved safety issue in 1978 after it became apparent that steam generator tubes were subject to widespread degradation, frequent leaks, and occasional ruptures (i.e., gross failures). USI Task Action Plans A-3, A-4, and A-5 were established to evaluate the safety significance of these problems for Westinghouse, Combustion Engineering, and Babcock & Wilcox steam generators, respectively. These studies were later combined into a single effort because PWR vendors were all experiencing many of the same problems.

NUREG-0844 provides a generic risk assessment that indicates that risk from steam generator tube rupture (SGTR) events is not a significant contributor to the total risk at a given site, nor to the total risk to which the general public is routinely exposed. This finding is considered indicative of the effectiveness of licensee programs and regulatory requirements for ensuring steam generator tube integrity in accordance with 10 CFR Part 50, Appendices A and B.

NUREG-0844 also identifies a number of staff-recommended actions that can further improve the effectiveness of licensee programs in ensuring the integrity of steam generator tubes and in mitigating the consequences of a SGTR. As part of the integrated program, the staff issued Generic Letter 85-02 encouraging licensees of PWRs to upgrade their programs, as necessary, to meet the intent of the staff-recommended actions; however, such recommended actions do not constitute NRC requirements. The staff's assessment of licensee responses to Generic Letter 85-02 was provided to the Commission in SECY 86-97.

4. USI NO. A-6 TITLE: Mark I Containment Short-Term Program

This USI was resolved in December 1977 with the publication of NUREG-0408, "Mark I Containment Short-Term Program Safety Evaluation Report."

The objectives of the Mark I short-term program were: (a) to examine the containment system of each BWR facility with a Mark I containment design to verify that it would maintain its integrity and functional capability when subjected to the most probable hydrodynamic loads induced by a postulated

ENCLOSURE 2

ISSUES SUMMARIES FOR UNRESOLVED SAFETY ISSUES

1. USI NO. A-1 TITLE: Water Hammer

This Unresolved Safety Issue (USI) was resolved in March 1984, with the publication of NUREG-0927, "Evaluation of Water Hammer in Nuclear Power Plants-Technical Findings Relevant to Unresolved Safety Issue A-1." Also on March 15, 1984, the EDO sent the Commissioners SECY 84-119 titled, "Resolution of Unresolved Safety Issue A-1, Water Hammer."

In SECY 84-119, the staff concluded that the frequency and severity of water hammer occurrences had been significantly reduced through (a) incorporation of design features such as keep-full systems, vacuum breakers, J-tubes, void detection systems, and improved venting procedures; (b) proper design of feedwater valves and control systems; and (c) increased operator awareness and training. Therefore, the resolution of USI A-1 did not involve any hardware or design changes on existing plants. It did involve Standard Review Plan (SRP) changes (forward fits) and a comprehensive set of guidelines and criteria to evaluate and upgrade utility training programs (per TMI Task Action Plan Item I.A.2.3). In addition, the assumption was made that for BWRs with isolation condensers (ICs) a reactor-vessel high water-level feedwater pump trip was in place or being installed. This was necessary because calculated values had postulated an IC failure by water hammer that opened a direct pathway to the environment.

2. USI NO. A-2 TITLE: Asymmetric Blowdown Loads in Reactor Coolant System

This USI was resolved in January 1981 with the publication of NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems."

In October 1975, the NRC notified each operating PWR licensee of a potential safety problem concerning the fact that asymmetric LOCA loads had not been considered in the design of any PWR piping system. In June 1976 the NRC informed each PWR licensee that it was required to reassess the reactor vessel support design of its facility. The staff expanded the scope of the problem in January 1978 with a request for additional information to all PWR licensees. NUREG-0609 provided guidance for these analyses. For operating PWRs, Multi-Plant Action (MPA) Item D-10 was established by NRC's Division of Licensing for implementation purposes.

During the course of the work on USI A-2, it was demonstrated that there were only a very limited number of break locations which could give rise to significant loads. Subsequently, after substantial new technical work, it was demonstrated that pipes would leak before break and that new fracture mechanics techniques for the analyzing of piping failures assured adequate protection against failures in primary system piping in PWRs (Generic Letter 84-04). This was reflected in a revision of General Design Criteria (GDC)-4 (Appendix A to 10 CFR Part 50) published in the Federal Register in final form on April 11, 1986, and in a subsequent revision to GDC-4 published in the Federal Register design-basis LOCA, and (b) to verify that licensed Mark I BWR facilities could continue to operate safely, without undue risk to the public health and safety until such time as a methodical, comprehensive long-term program is conducted.

The NRC staff used a safety factor of at least two to failure for the weakest structural or mechanical component in the Mark I containment system in judging that containment integrity and functions would be assured under most probable design-basis LOCA-induced hydrodynamic loads.

As indicated in NUREG-0408, the staff required full implementation of the calculation of the hydrodynamic loads and structural analysis as an interim measure until complete implementation of the long-term program had been achieved. In NUREG-0408 the staff concluded that the objectives of the Short-Term Program had been satisfied, thus documenting the basis for resolving this safety issue. This issue is considered complete for all affected BWRs.

5. USI NO. A-7 TITLE: Mark I Long-Term Program

This USI was resolved in August 1982 with the publication of Supplement 1 to NUREG-0661, "Safety Evaluation Report, Mark I Containment Long-Term Program" and Standard Review Plan Section 6.2.1.1.C. For operating BWRs, MPA D-01 was established for implementation purposes.

The focus of this USI was the suppression pool hydrodynamic loads, associated with a postulated LOCA, which had not explicitly been included in the original Mark I containment design. The issue was identified during large-scale testing of a Mark III containment design. The staff addressed this issue in NUREG-0661, published in July 1980, and in Supplement 1 to NUREG-0661, published in August 1982.

The objective of the long-term program (LTP) was to establish the design-basis loads that are appropriate for the anticipated life of each Mark I BWR facility and to restore the originally intended design-safety margins for each Mark I containment system. The principal thrust of the LTP was the development of generic methods for defining suppression pool hydrodynamic loadings and the associated structural assessment techniques for the Mark I configuration. On the basis of experimental and analytical programs conducted by the Mark I Owners Group, it was determined that the hydrodynamic load definition procedures, with some modifications defined in NUREG-0661, provided a conservative estimate of these loading conditions. Thus, the requirements associated with this USI were concerned with the structural assessment of Mark I containments and related structures to the hydrodynamic loads defined by the staff in the LTP.

In January 1981, the staff issued "Orders For Modification of License and Grant of Extension of Exemptions" to each licensee of a Mark I plant. The orders required the licensees to assess the suppression pool hydrodynamic loads in accordance with General Electric documents and NUREG-0661 on a defined schedule. For some plants, the implementation schedule was extended by a subsequent order.

6. USI NO. A-8 TITLE: Mark II Containment Pool Dynamic Loads

This USI was resolved in August 1981 with the publication of NUREG-0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria," and Standard Review Plan (SRP) Section 6.2.1.1C. The requirement is that the 11 BWRs having the Mark II containment shall meet the requirements of GDC 16.

As stated in NUREG-0808, the original design of the Mark II containment system considered only those loads normally associated with design-basis accidents that were known at the time. These included pressure and temperature loads associated with a LOCA, seismic loads, dead loads, jet impingement loads, hydrostatic loads due to water in the suppression chamber, overload pressure test loads, and construction loads. However, since the establishment of the original design criteria, additional loading conditions were identified that must be considered for the pressure-suppression containment-system design.

In the course of performing large-scale testing of an advanced design pressure-suppression containment (Mark III), and during inplant testing of Mark I containments, new suppression-pool hydrodynamic loads were identified that had not been included explicitly in the original Mark II containment-design basis. These additional loads result from dynamic effects of drywell air and steam being rapidly forced into the suppression pool during a postulated LOCA and from suppression-pool response to safety/relief valve (SRV) operation; these are generally associated with plant transient operating conditions. Because these new hydrodynamic loads had not been considered, the NRC staff determined that a detailed reevaluation of the Mark II containment system was required.

The issuance of NUREG-0808, NUREG-0802, "Safety Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments," and NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," documented acceptable methods for calculating the hydrodymanic loads associated with plant transient conditions. Specifically, the loads referenced in these NRC staff reports, as modified by the acceptance criteria, constituted the resolution of USI A-8. SRP Section 6.2.1 has been modified to reflect the applicability of these reports to Mark II containment evaluations.

Implementation is believed to be complete for all Mark II BWRS. As part of the licensing process, the staff required that the applicants utilize the new calculation methodology defined in the reference documents before a full power license was issued.

7. USI NO. A-9 TITLE: Anticipated Transient Without Scram (ATWS) per 10 CFR 50.62

ISSUES SUMMARY:

This USI was resolved in June 1984 with the publication of a final rule (10 CFR 50.62) to require improvements in plants to reduce the likelihood of failure of the reactor protection system (RPS) to shut down the reactor following anticipated transients and to mitigate the consequences of an anticipated transient without scram (ATWS) event.

The rule includes the following design-related requirements: 50.62(C)(1), diverse and independent auxiliary feedwater initiation and turbine trip for all PWRs; 50.62(C)(2), diverse scram systems for CE and B&W reactors; 50.62(C)(3) alternate rod injection (ARI) for BWRs; 50.62(C)(4); standby liquid control system (SLCS) for BWRs; and 50.62(C)(5), automatic trip of recirculation pumps under conditions indicative of an ATWS for BWRs. Information requirements and an implementation schedule are also specified.

8. USI NO. A-10 TITLE: BWR Feedwater Nozzle Cracking

This issue was resolved in November 1980 with the publication of NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking." MPA B-25 was established by NRC's Division of Licensing for implementation purposes.

Inspections of operating BWRs conducted up to April 1978 revealed cracks in the feedwater nozzles of 20 reactor vessels. It was determined that cracking was due to high-cycle fatigue caused by fluctuations in water temperature within the vessel in the nozzle region.

By letter dated November 13, 1980, Darrell G. Eisenhut provided licensees with a copy of NUREG-0619. The letter stated that NUREG-0619 provided the resolution of the staff's generic technical activity USI A-10, which resulted from the inservice discovery of cracking in feedwater nozzles and control rod drive return line nozzles. NUREG-0619 describes the technical issues, General Electric and staff studies and analyses, and the staff's positions and requirements. Licensees were required to respond, pursuant to 10 CFR 50.54(f), that they would meet implementation dates indicated in NUREG-0619.

Generic Letter 81-11 was subsequently issued to provide technical clarification to the November 13, 1980 letter, to clarify that it had been sent to PWR licensees for information only, and that no response was required from PWR licensees.

9. USI NO. A-11 TITLE: Reactor Vessel Materials Toughness

This USI was resolved in October 1982 with the publication of NUREG-0744, "Pressure Vessel Material Fracture Toughness.". NUREG-0744 was issued by Generic Letter 82-26 and provided only a methodology to satisfy the requirements of 10 CFR Part 50, Appendix G. No licensee response to Generic Letter 82-26 was required.

Because of the remote possibility that nuclear reactor pressure vessels designed to the ASME Boiler and Pressure Vessel Code would fail, the design of nuclear facilities does not provide protection against reactor vessel failure. Prevention of reactor vessel failure depends primarily on maintaining the reactor vessel material fracture toughness at levels that will resist brittle fracture during plant operation. At service times and operating conditions typical of current operating plants, reactor vessel fracture toughness properties provide adequate margins of safety against vessel failure; however, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial safety margins.

Appendix G to 10 CFR Part 50 requires that the Charpy upper shelf energy throughout the life of the vessel be no less than 50 ft-lb unless it is demonstrated that lower values will provide margins of safety against failure equivalent to those provided by Appendix G of the ASME code. USI A-11 was initiated to address the staff's concern that some vessels were projected to have beltline materials with Charpy upper shelf energy less than 50 ft-lb.

NUREG-0744 provides a method for evaluating reactor vessel materials when their Charpy upper shelf energy is predicted to fall below 50 ft-lb. Plants will use the prescribed method when analysis of irradiation damage predicts that the charpy upper shelf energy is below 50 ft-lb.

10. USI NO. A-12 TITLE: Potential of Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports

This USI was resolved in October 1983 with the publication of NUREG-0577, "Potential of Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports." The resolution contained no backfit requirements; it only applied to plants with a new construction permit issued after October 1983. Standard Review Plan Section 5.3.4 was issued at the same time this USI was resolved.

The concern in this USI, as the title indicates, was the potential of low fracture toughness of some materials selected for fabrication of steam generator (SG) and reactor coolant pump (RCP) supports in operating PWRs. Lamellar tearing was also of concern. Fracture toughness is a measure of a material's resistance to fracture in the presence of a previously existing crack. Generally, a material is considered to have adequate fracture toughness if it can withstand loading to its design limit in the presence of detectable flaws under stated conditions of stress and temperature.

The modifications to address this USI could involve maintaining minimum temperature around the supports above its fracture transition temperature, or total replacement of existing SG and RCP supports with supports fabricated of material grade which has a higher Charpy upper shelf energy and a lower transition temperature. Analysis performed for the resolution of this USI determined that, even with the failure of the SG and RCP supports, the amount of incremental release of radioactivity would not be sufficiently high enough to justify any modification in terms of increasing the toughness of these supports. This conclusion is based on a value-impact analysis documented in Appendix C of NUREG-0577.

11. USI NO. A-17 TITLE: Systems Interactions in Nuclear Power Plants

Generic Letter (GL) 89-18, dated September 6, 1989, was sent to all power reactor licensees and constitutes the resolution of USI A-17. The generic letter did not require any licensee actions.

GL 89-18 had two enclosures which (a) outlined the bases for the resolution of USI A-17, and (b) provided five general lessons learned from the review of the overall systems interaction issue. The staff anticipated that licensees would review this information in other programs, such as the Individual Plant Examination (IPE) for Severe Accident Vulnerabilities. Specifically, the staff expected that insights concerning water intrusion and flooding from internal sources, as described in the appendix to NUREG-1174, would be considered in the IPE program. Also considered in the resolution of this USI was the expectation that licensees would continue to review information on events at operating nuclear power plants in accordance with the requirements of TMI Task Action Plan Item I.C.5 (NUREG-0737).

12. USI NO. A-24 TITLE: Qualification of Class 1E Equipment

This USI was resolved in July 1981 with the publication of NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." Part I of the report is the original NUREG-0588 that was issued for comment; that report, in conjunction with the Division of Operating Reactor (DOR) Guidelines, was endorsed by a Commission Memorandum and Order as the interim position on this subject until "final" positions were established in rule making. On January 21, 1983 the Commission amended 10 CFR 50.49 (the rule), effective February 22, 1983, to codify existing qualification methods in national standards, regulatory guides, and certain NRC publications, including NUREG-0588.

The rule is based on the DOR Guidelines and NUREG-0588. These provide guidance on (a) how to establish environmental service conditions, (b) how to select methods which are considered appropriate for qualifying the equipment in different areas of the plant, and (c) such other areas as margin, aging, and documentation. NUREG-0588 does not address all areas of qualification; it does supplement, in selected areas, the provisions of the 1971 and 1974 versions of IEEE Standard 323. The rule recognizes previous qualification efforts completed as a result of Commission Memorandum and Order CLI-80-21 and also reflects different versions IEEE 323, dependent on the date of the construction permit Safety Evaluation Report (SER). Therefore, plant-specific requirements may vary in accordance with the rule.

In summary, the resolution of A-24 is embodied in 10 CFR 50.49. A measure of whether each licensee has implemented the resolution of A-24 may therefore be found in the determination of compliance with 10 CFR 50.49. This was addressed by 72 SERs for operating plants issued shortly after publication of the rule and subsequently in operating license reviews pursuant to Standard Review Plan Section 3.11. This was further addressed by the first-round environmental qualification inspections conducted by the NRC.

13. USI NO. A-26 TITLE: Reactor Vessel Pressure Transient Protection

This USI was resolved in September 1978 with the publication of NUREG-0224, "Reactor Vessel Pressure Transient Protection for PWRs," and Standard Review Plan Section 5.2. The licensees of all operating PWRs were requested to provide an overpressure prevention system that could be used whenever the plants were in startup or shutdown conditions. The issue affected all operating and future plants, and the staff established MPA B-04 for implementing the solution at operating PWRs.

Since 1972, there have been numerous reported incidents of pressure transients in PWRs where technical specification pressure and temperature limits have been exceeded. The majority of these events occurred while the reactors were in a solid-water condition during startup or shutdown and at relatively low reactor vessel temperatures. Since the reactor vessels have less toughness at lower temperatures, they are more susceptible to brittle fracture under these conditions than at normal operating temperatures. In light of the frequency of the reported transients and the associated potential for vessel damage, the NRC staff concluded that measures should be taken to minimize the number of future transients and reduce their severity.

Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," was published July 12, 1988. This generic letter provides guidance regarding review of pressure-temperature limits and indicates that licensees may have to revise low-temperature-overpressure protection setpoints.

14. USI NO. <u>A-31</u> TITLE: <u>Residual Heat Removal Shutdown Requirements</u>

This USI was resolved in May 1978 with the publication of Standard Review Plan (SRP) Section 5.4.7. Only those plants expected to receive an operating license after January 1, 1979 were affected by this resolution. The USI involved establishment of criteria for the design and operation of systems necessary to take a power reactor from normal operating conditions to cold shutdown.

SRP Section 5.4.7 stated that, for purposes of implementation, plants would be divided into three classes: Class 1 would require full compliance for Construction Permit (CP) or Preliminary Design Approval (PDA) applications which were docketed on or after January 1, 1978. Class 2 required a partial implementation for all plants for which CP or PDA applications were docketed before January 1, 1978, and for which an Operating License (OL) issuance was expected on or after January 1, 1979. Class 3 affected all operating reactors and all other plants for which issuance of the OL was expected before January 1, 1979. The extent to which Class 3 plants would require implementation was based on the combined staff review of related plant features. In general, the outcome of these evaluations were that only plants receiving an OL after January 1, 1979 were affected by this USI resolution, and there were no backfits to operating plants that had received an operating license before January 1, 1979.

15. USI NO. <u>A-36</u> TITLE: <u>Control of Heavy Loads, Phases I & II</u>

This USI was resolved in July 1980 with the publication of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and Standard Review Plan (SRP) Section 9.1.5. The staff established MPAs C-10 and C-15 for the implementation of Phases I and II, respectively, of the resolution of this issue at operating plants.

In nuclear power plants, heavy loads may be handled in several plant areas. If these loads were to drop in certain locations in the plant, they may impact spent fuel, fuel in the core, or equipment that may be required to achieve safe shutdown and continue decay heat removal. USI A-36 was established to systematically examine staff licensing criteria and the adequacy of measures in effect at operating plants, and to recommend necessary changes to ensure the safe handling of heavy loads. The guidelines proposed in NUREG-0612 include definition of safe load paths, use of load handling procedures, training of crane operators, guidelines on slings and special lifting devices, periodic inspection and maintenance for the crane, as well as various alternatives.

By Generic Letters dated December 22, 1980, and February 3, 1981 (Generic Letter 81-07), all utilities were requested to evaluate their plants against the guidance of NUREG-0612 and to provide their submittals in two parts: Phase I (six month response) and Phase II (nine month response). Phase I responses were to address Section 5.1.1 of NUREG-0612 which covered the following areas:

1. Definition of safe load paths
2. Development of load handling procedures
3. Periodic inspection and testing of cranes
4. Qualifications, training and specified conduct of operators
5. Special lifting devices should satisfy the guidelines of ANSI N14.6.6.
6. Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9
7. Design of cranes to ANSI B30.2 or CMAA-70

Phase II responses were to address Sections 5.1.2 thru 5.1.6 of NUREG-0612 which covered the need for electrical interlocks/mechanical stops, or alternatively, single-failure-proof cranes or load drop analyses in the spent fuel pool area (PWR), containment building (PWR), reactor building (BWR), other areas and the specific guidelines for single-failure-proof handling systems.

As stated in Generic Letter 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants' - NUREG-0612," all licensees have completed the requirement to perform a review and submit a Phase I and a Phase II report. Based on the improvements in heavy loads handling obtained from implementation of NUREG-0612 (Phase I), further action was not required to reduce the risks associated with the handling of heavy loads. Therefore, a detailed Phase II review of heavy loads was not necessary and Phase II was considered completed.

While not a requirement, NRC encouraged the implementation of any actions identified in Phase II regarding the handling of heavy loads that were considered appropriate.

16. USI NO. <u>A-39</u> TITLE: <u>Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits</u>

This USI was resolved with the publication of Standard Review Plan (SRP) Section 6.2.1.1.C, in October 1982. In addition, NUREGs 0763, 0783 and 0802 were issued for Mark I, Mark II, and Mark III containments, respectively.

BWR plants are equipped with safety/relief valves (SRVs) to protect the reactor from overpressurization. Plant operational transients, such as turbine trips, will actuate the SRV. Once the SRV opens, the air column within the partially submerged discharge line is compressed by the high-pressure steam released from the reactor. The compressed air discharged into the suppression pool produces high-pressure bubbles. Oscillatory expansion and contraction of these bubbles create hydrodynamic loads on the containment structures, piping, and equipment inside containment.

NUREG-0802 presents the results of the staff's evaluation of SRV loads. The evaluation, however, is limited to the quencher devices used in Mark II and III containments. With respect to Mark I containments, the SRV acceptance criteria are presented in NUREG-0661, "Safety Evaluation Report, Mark I Containment and Long-Term Program," and are dealt with as part of USI A-7.

SRP Section 6.2.1.1.C addresses the applicable review criteria, since all Mark II and III containment designs are understood to have completed their operating license (OL) reviews subsequent to resolution of this USI and reflection of the resolution in the SRP.

17. USI NO. A-40 TITLE: Seismic Design Criteria

The staff has resolved USI A-40 as documented in NUREG/CR-5347, "Recommendations for Resolution of Public Comments on USI A-40," issued in June 1989, and NUREG-1233, "Regulatory Analysis for USI A-40," issued in September 1989.

For plants not covered under the scope of USI A-46, "Seismic Qualification of Equipment in Operating Plants," the staff concluded that tanks in plants that were subject to licensing review by the staff after 1984 had been reviewed to current requirements and found acceptable. For tanks in plants reviewed during 1980-1984, the staff identified four plant sites (six units) that were not explicitly reviewed to current requirements. The four plants (Callaway 1/2, Wolf Creek, Shearon Harris 1, and Watts Bar 1/2) are being handled on a plant-specific basis.

USI A-40 originated in 1977. The basic objectives were (a) to study the seismic design criteria, (b) to quantify the conservatism associated with the criteria, and (c) to recommend modifications to the Standard Review Plan (SRP) if changes are justified. Lawrence Livermore National Laboratory (LLNL) completed the study and published its findings in NUREG/CR-1161, "Recommended Revisions to USNRC - Seismic Design Criteria," dated May 1980. The report recommended specific changes to the Standard Review Plan (SRP). NRC staff reviewed the report and developed some other changes that would reflect the present state of seismic design practices. The resulting SRP changes were issued for public comment in June 1988, and the final SRP changes are to be published in October 1989.

The major SRP changes consist of (a) clarification of development of site specific spectra, (b) justification for use of single synthetic time-history by power spectral density function, (c) location and reductions of input ground motion for soil structure interaction, and (d) design of above-ground vertical tanks. Except for item (d), these items do not constitute any additional requirements for current licenses and applications, and thus, no backfitting is being required for these items. However, the revised provisions could be used for margin studies and reevaluations or individual plant examination for external events (IPEEE).

The participant utilities in the Seismic Qualification Utility Group (SQUG) agreed to implement the changed criteria for flexible vertical tanks for their plants. For the four plants where this issue has to be resolved on an individual basis a 10 CFR 50.54(f) request-for-information letter has been sent to the affected utilities. If the information received indicates that large above-ground vertical tanks do not meet the new criteria, plant-specific backfits will be considered.

18. USI NO. A-42 TITLE: Pipe Cracks in Boiling Water Reactors

This USI was resolved in February 1981 with the publication of NUREG-0313, Revision 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping." That NUREG document was issued to all holders of BWR operating licenses or construction permits and to all applicants for BWR operating licenses. The staff established MPA B-05 for implementation of the resolution at operating plants.

Pipes have cracked in the heat-affected zones of welds in primary system piping in BWRs since mid-1960. These cracks have occurred mainly in Type 304 stainless steel, which is the type used in most operating BWRs. The major problem is recognized to be intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel components that have been made susceptible to this failure by being "sensitized," either by post-weld heat treatment or by sensitization of a narrow heat affected zone near welds.

"Safe ends" that have been highly sensitized by furnace heat treatment while attached to vessels during fabrication were found to be susceptible to IGSCC in the late 1960s. Most of the furnace-sensitized safe ends in older plants have been removed or clad with a protective material, and only a few BWRs still have furnace-sensitized safe ends in use. Most of these, however, are in smaller diameter lines.

Cracks reported before 1975 occurred primarily in 4-inch-diameter recirculation loop bypass lines and in 10-inch-diameter core spray lines. Cracking is most often detected during inservice inspections using ultrasonic test techniques. Some piping cracks have been discovered as a result of primary coolant leaks.

NUREG-0313, Revision 1 provided the NRC staff's revised acceptable methods for reducing the IGSCC susceptibility of BWR code class 1, 2, and 3 pressure boundary piping of sizes identified above and safe ends. In addition, it provided the requirements for augmented inservice inspection of piping with nonconforming materials.

As a result of further IGSCC degradations in larger piping, the staff provided licensees with additional requirements in several NRC communications (i.e., Bulletins 82-03, 83-2, and 84-11). The long-term resolution of IGSCC in BWR piping (including the scope of A-42) was provided in NUREG-0313, Revision 2 which was transmitted to all holders of BWR operating licenses via Generic Letter 88-01.

19. USI NO. A-43 TITLE: Containment Emergency Sump Performance

The resolution of this USI was presented to the Commission in October 1985 in SECY-85-349. NUREG-0897, Revision 1, "Containment Emergency Sump Performance," presents the results of the staff's technical findings. These findings established a need to revise current licensing guidance on these matters. RG 1.82 Revision 0 and Standard Review Plan Section 6.2.2, "Containment Heat Removal Systems" were revised to reflect this new guidance. No licensee actions were required.

Initially, an issue existed concerning the availability of adequate recirculation cooling water following a loss-of-coolant accident (LOCA) when long-term recirculation of cooling water from the PWR containment sump, or the BWR residual heat removal system (RHR) suction intake, must be initiated and maintained to prevent core melt.

The technical concerns evaluated under USI A-43 were: (a) post-LOCA adverse conditions resulting from potential vortex formation and air ingestion and subsequent pump failure, (b) blockage of sump screens with LOCA generated insulation debris causing inadequate net positive suction head (NPSH) on pumps, and (c) RHR and containment spray pumps inoperability due to possible air, debris, or particulate ingestion on pump seal and bearing systems.

This revised guidance applies only to future construction permits, preliminary design approvals, final design approvals, standardized designs, and applications for licenses to manufacture. The staff performed a regulatory analysis to determine if this new guidance should be applied to operating plants. The results of this analysis were reported in NUREG-0869 Revision 1, "USI A-43 Regulatory Analysis," issued in October 1985. The staff concluded that the regulatory analysis does not support any new generic requirements for present licensees to perform debris assessments.

20. USI NO. A-44 TITLE: Station Blackout

This USI was resolved in June 1988 with the publication of a new rule (10 CFR 50.63) and Regulatory Guide 1.155.

Station blackout means the loss of offsite ac power to the essential and nonessential electrical buses concurrent with turbine trip and the unavailability of the redundant onsite emergency ac power systems. WASH-1400 showed that station blackout could be an important risk contributor, and operating experience has indicated that the reliability of ac power systems might be less than originally anticipated. For these reasons station blackout was designated as a USI in 1980. A proposed rule was published for comment on March 21, 1986. A final rule, 10 CFR 50.63, was published on June 21, 1988 and became effective on July 21, 1988. Regulatory Guide 1.155 was issued at the same time as the rule and references an industry guidance document, NUMARC-8700. In order to comply with the A-44 resolution, licensees will be required to:

- maintain onsite emergency ac power supply reliability above a minimum level
- develop procedures and training for recovery from a station blackout
- determine the duration of a station blackout that the plant should be able to withstand
- use an alternate qualified ac power source, if available, to cope with a station blackout
- evaluate the plant's actual capability to withstand and recover from a station blackout
- backfit hardware modifications if necessary to improve coping ability

Section 50.63(c)(1) of the rule required each licensee to submit a response including the results of a coping analysis within 270 days from issuance of an operating license or the effective date of the rule, whichever is later.

21. USI NO. A-45 TITLE: Shutdown Decay Heat Removal Requirements

USI A-45 was resolved by SECY 88-260, "Shutdown Decay Heat Removal Requirements (USI-A-45)," issued September 13, 1988, without imposing any new licensing requirements other than the Individual Plant Examination (IPE), as described below. At the same time the staff issued NUREG-1289, "Regulatory and Backfit Analysis: USI A-45." Since all of the significant USI A-45 results have been found to be highly plant specific, the Commission decided it was not appropriate to propose a single generic corrective action to be applied uniformly to all plants.

The Commission is currently implementing the Severe Accident Policy (50 FR 32138) and will require all plants presently operating or under construction to undergo a systematic examination termed the IPE. The reason for this examination is to identify any plant-specific vulnerabilities to severe accidents. The IPE analysis intends to examine and understand the plant emergency procedures, design, operations, maintenance, and surveillance, in order to identify vulnerabilities. The analysis will examine both the decay heat removal systems and those systems used for other related functions. This includes CE plants without power-operated relief valves.

NRC has decided to subsume A-45 into the IPE program as the most effective way of achieving resolution of specific plant concerns associated with A-45.

22. USI NO. A-46 TITLE: Seismic Qualification of Equipment in Operating Plants

USI A-46 was resolved with the issuance of GL 87-02 on February 19, 1987, which endorsed the approach of using the seismic and test experience data proposed by the Seismic Qualification Utility Group (SQUG) and Electric Power Research Institute (EPRI). This approach was endorsed by the Senior Seismic Review and Advisory Panel (SSRAP) and approved by the NRC staff.

The scope of the review was narrowed to equipment required to bring each affected plant to hot shutdown and maintain it there for a minimum of 72 hours. The review includes a walkthrough of each plant which is required to inspect equipment. Evaluation of equipment will include: (a) adequacy of equipment anchorage; (b) functional capability of essential relays; (c) outliers and deficiencies (i.e., equipment with non-standard configurations); and (d) seismic systems interation.

As an outgrowth of the Systematic Evaluation Program (SEP), the need was identified for reassessing design criteria and methods for the seismic qualification of mechanical equipment and electrical equipment. Therefore, the seismic qualification of the equipment in operating plants must be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this issue was to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at operating plants in lieu of attempting to backfit current design criteria for new plants.

Generic Letter 87-02 with associated guidance, required all affected utilities to evaluate the seismic adequacy of their plants. The specific requirements and approach for implementation are being developed jointly by SQUG and the staff on a generic basis before individual member utilities proceed with plant-specific implementation.

23. USI NO. A-47 TITLE: Safety Implication of Control Systems in LWR Nuclear Power Plants

USI A-47 was resolved September 20, 1989, with the publication of Generic Letter (GL) 88-19.

The generic letter states:

> "The staff has concluded that all PWR plants should provide automatic steam generator overfill protection, all BWR plants should provide automatic reactor vessel overfill protection, and that plant procedures and technical specifications for all plants should include provisions to verify periodically the operability of the overfill protection and to assure that automatic overfill protection is available to mitigate main feedwater overfeed events during reactor power operation. Also, the system design and setpoints should be selected with the objective of minimizing inadvertent trips of the main feedwater system during plant startup, normal operation, and protection system surveillance. The Technical Specifications recommendations are consistent with the criteria and the risk considerations of the Commission Interim Policy Statement on Technical Specification Improvement. In addition, the staff recommends that all BWR recipients reassess and modify, if needed, their

operating procedures and operator training to assure that the operators can mitigate reactor vessel overfill events that may occur via the condensate booster pumps during reduced system pressure operation."

Also, page 2 of the generic letter provides for additional actions for CE and B&W plants. The generic letter provides amplifying guidance for licensees.

The generic letter requires that licensees provide NRC with their schedule and commitments within 180 days of the letter's date. The implementation schedule for actions on which commitments are made should be prior to startup after the first refueling outage, but no later than the second refueling outage, beginning 9 months after receipt of the letter.

24. USI NO. A-48 TITLE: Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

The NRC staff concluded April 19, 1989, that USI A-48 is resolved, as stated in SECY 89-122.

USI A-48 was initiated as a result of the large amount of hydrogen generated and burned within containment during the Three Mile Island (TMI) accident. This issue covers hydrogen control measures for recoverable degraded core accidents for all BWRs and those PWRs with ice condenser containments. Extensive research in this area has led to significant revision of the Commission's hydrogen control regulations, given in 10 CFR 50.44, published December 2, 1981.

10 CFR 50.44 requires inerting of BWR Mark I and Mark II containments as a method for hydrogen control. The BWR Mark I and Mark II reactor containments have operated for a number of years with an inerted atmosphere (by addition of an inert gas, such as nitrogen) which effectively precludes combustion of any hydrogen generated. USI A-48 with respect to BWR Mark I and II containments is not only resolved but understood to be fully implemented in the affected plants.

The rule for BWRs with Mark III containments and PWRs with ice condenser containments was published on January 25, 1985. The rule required that these plants be provided with a means for controlling the quantity of hydrogen produced, but did not specify the control method. In addition, the task action plan for USI A-48 provided for plant-specific reviews of lead plants for reactors with Mark III and ice condenser containments. Sequoyah was chosen as the lead plant for ice condenser containments and Grand Gulf for Mark III containments. Both of the lead plant licensees chose to install igniter-type systems which would burn the hydrogen before it reached threatening concentrations within the containment. Final design igniter systems have been installed not only in both lead plants, Sequoyah and Grand Gulf, but in all other ice condenser and Mark III plants as well. The staff's safety evaluations of the final analyses required to be submitted by these licensees by the rule are scheduled for completion in 1989.

Large dry PWR containments were excluded from USI A-48 because they have a greater ability to accommodate the large quantities of hydrogen associated with a recoverable degraded core accident than the smaller Mark I, II, III and ice condenser containments. However, this issue has continued to be considered and, in 1989, hydrogen control for large dry PWR containments was identified as a high-priority Generic Issue (GI) 121. The resolution of GI 121 is being actively pursued in close coordination with more recent research findings.

25. USI NO. A-49 TITLE: Pressurized Thermal Shock

The final rule (10 CFR 50.61) on pressurized thermal shock (PTS) was approved by the Commission in July 1985. Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for PWRs," was later published in February 1987. Thus, this issue was resolved and new requirements were established, applicable to PWRs only. The rule required that each operating reactor meet the screening criteria provided in the rule or provide supplemental analysis to demonstrate that PTS is not a concern for the facility.

Neutron irradiation of reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. The fracture toughness sensitivity to radiation-induced change is increased by the presence of certain materials such as copper. Decreased fracture toughness makes it more likely that, if a severe overcooling event occurs followed by or concurrent with high vessel pressure, and if a small crack is present on the vessel's inner surface, that crack could grow to a size that might threaten vessel integrity.

Severe pressurized overcooling events are improbable since they require multiple failures and improper operator performance. However, certain precursor events have happened that could have potentially threatened vessel integrity if additional failures had occurred and/or if the vessel had been more highly irradiated. Therefore, the possibility of vessel failure due to a severe pressurized overcooling event cannot be ruled out.

LIST OF RECENTLY ISSUED GENERIC LETTERS

Generic Letter No.	Subject	Date of Issuance	Issued To
89-21	REQUEST FOR INFORMATION CONCERNING STATUS OF IMPLEMENTATION OF UNRESOLVED SAFETY ISSUE (USI) REQUIREMENTS	10/19/89	ALL HOLDERS OF OPERATING LICENSES AND CONSTRUCTION PERMITS FOR NUCLEAR POWER REACTORS
89-20	PROTECTED AREA LONG-TERM HOUSEKEEPING	09/26/89	ALL FUEL CYCLE FACILITY LICENSEES WHO POSSESS, USE, OR PROCESS FORMULA QUANTITIES OF STRATEGIC SPECIAL NUCLEAR MATERIAL
89-19	REQUEST FOR ACTION RELATED TO RESOLUTION OF UNRESOLVED SAFETY ISSUE A-47 "SAFETY IMPLICATION OF CONTROL SYSTEMS IN LWR NUCLEAR POWER PLANTS" PURSUANT TO 10 CFR 50.54(f)	09/20/89	ALL LICENSEES OF OPERATING REACTORS, APPLICANTS FOR OPERATING LICENSES AND HOLDERS OF CONSTRUCTION PERMITS FOR LIGHT WATER REACTOR NUCLEAR POWER PLANTS
89-18	RESOLUTION OF UNRESOLVED SAFETY ISSUE A-17, "SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS ACCESSION NUMBER IS 8909070029	09/06/89	ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR NUCLEAR POWER PLANTS
89-17	PLANNED ADMINISTRATIVE CHANGES TO THE NRC OPERATOR LICENSING WRITTEN EXAMINATION PROCESS - GENERIC LETTER 89-17	09/06/89	ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR PWRS AND BWRS AND ALL LICENSED OPERATORS
89-16	INSTALLATION OF A HARDENED WETWELL VENT (GENERIC LETTER 89-16)	09/01/89	ALL GE PLANTS
88-20 SUPPLEMENT 1	GENERIC LETTER 88-20 SUPPLEMENT NO. 1 (INITIATION OF THE INDIVIDUAL PLANT EXAMINATION FOR SEVERE VULNERABILITIES 10 CFR 50.54(f))	08/29/89	ALL LICENSEES HOLDING OPERATING LICENSES AND CONSTRUCTION PERMITS FOR NUCLEAR POWER REACTOR FACILITIES

This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires December 31, 1989. The estimated average burden hours is 80 person hours per plant, including searching data sources, gathering and analyzing the data, and preparing the required letter. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Records and Reports Management Branch, Division of Information Support Services, Office of Information Resources Management, 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.

Sincerely,

Original signed by
James G. Partlow

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

Enclosures:
1. USI Table
2. USI Issues Summary
3. List of Most Recently Issued NRC Generic Letters

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10/13/89	10/13/89	10/13/89	10/12/89	10/13/89
OGCB:NRR*	D:DRP*	ADP:NRR		
CBerlinger	SVarga	JPartlow		
10/12/89	10/13/89	10/16/89		

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Sincerely,

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

Enclosure:
1. USI Table
2. USI Issues Summary

KTE TA:ADR2:DRP KEccleston 10/13/89	TSB:NRR BBarber 10/13/89	D:PDI-4 RWessman 10/13/89	JTC EAD:NRR JRichardson 10/12/89	AT D:DST ~~SAD~~:NRR AThadani 10/13/89
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Technical questions should be directed to technical contacts identified on data sheets. Questions may also be directed to R. Hernan, G. Barber, K. Eccleston or R. Wessman.

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

cc: J. Taylor
H. Thompson
J. H. Sniezek
Division Directors, NRR
Associate Directors, NRR
Project Directors, NRR
Regional Administrators
J. Conran, CRGR
C. Berlinger, DOEA
S. Treby, OGC
A. J. Mediola
R. Dudley
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James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

cc: F. J. Miraglia
J. H. Sniezek
J. E. Richardson
L. C. Shao
A. C. Thadani
C. H. Berlinger
A. J. Mendiola
R. Dudley
M. Caruso
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Enclosure 2

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