Enclosure 1

CP&L

Carolina Power & Light Company

SERIAL: NLS-89-307

NOV 2 7 1989

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63

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

Gentlemen:

The information attached is provided in response to your Generic Letter 89-21 dated October 19, 1989 regarding the status of implementation of Unresolved Safety Issue (USI) requirements. Please note that the time provided to determine the status of implementation of all USIs as required by your request has been short. Therefore, the attached notations regarding the status of individual items are necessarily brief and cannot address all of the details of the more complex issues. The attached status has been compiled based upon a review of correspondence between CP&L and the NRC such as Safety Evaluation Reports (SER) and Inspection Reports, as well as other plant-specific documents.

If you have any questions concerning this information, please contact Mr. R. W. Prunty at (919)546-7318.

Yours very truly,

L. I. Loflin
Manager
Nuclear Licensing Section

LIL/LSR/lbf (538CRS)
Attachment

cc: Mr. R. A. Becker
Mr. S. D. Ebneter
Mr. J. E. Tedrow

41pp.

A017

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	NC	See Note 1
A-2/ MPA D-10	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	NUREG-0609 GL 84-04, GDC-4	PWR	C / (11/83)	See Note 2
A-3	Westinghouse Steam Generator Tube Integrity	NUREG-0844 SECY 86-97 SECY 88-272 GL 85-02 (No requirements)	W-PWR	C / (1/87)	See Note 3
A-4	CE Steam Generator Tube Integrity	NUREG-0844, SECY 86-97 SECY 88-272 GL 85-02 (No requirements)	CE-PWR	NA	
A-5	B&W Steam Generator Tube Integrity	NUREG-0844, SECY 86-97 SECY 88-272 GL 85-02 (No Requirements)	B&W-PWR	NA	
E A-6	Mark I Containment Short-Term Program	NUREG-0408	Mark I-BWR	NA	

* C - COMPLETE
NC - NO CHANGES NECESSARY
NA - NOT APPLICABLE
I - INCOMPLETE

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	NA	
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	NA	
A-9	Anticipated Transients Without Scram	NUREG-0460, Vol. 4 10 CFR 50.62	All	C / (7/89)	See Note 4
A-10/ MPA B-25	BWR Feedwater Nozzle Cracking	NUREG-0619 Letter from DG Eisenhut dated 11/13/80 GL 81-11	BWR	NA	
A-11	Reactor Vessel Material Toughness	NUREG-0744, Rev. 1 10 CFR 50.60/ 82-26	All	NA	See Note 5
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	NUREG-0577, Rev. 1 SRP Revision 5.3.4	PWR	NC	See Note 6
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	NC	See Note 7
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	All	C / (10/86)	See Note 8

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	I	See Note 9
A-31	Residual Heat Removal Shutdown Requirements	NUREG-0606 RG 1.113, RG 1.139 SRP 5.4.7	All OLs After 01/79.	C / (9/88)	See Note 10
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	C / (10/86)	See Note 11
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	NA	
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	C / (10/89)	See Note 12
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	NA	

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	NA	See Note 13
A-44	Station Blackout	RG 1.155 NUREG-1032 NUREG-1109 10 CFR 50.63	All	I	See Note 14
A-45	Shutdown Decay Heat Removal Requirements	SECY 88-260 NUREG-1289 NUREG/CR-5230 SECY 88-260 (No requirements)	All	I / (8/92)	See Note 15
A-46	Seismic Qualification of Equipment in Operating Plants	NUREG-1030 NUREG-1211/ GL 87-02, GL 87-03	All	NA	See Note 1[illegible]
A-47	Safety Implication of Control Systems	NUREG-1217, NUREG-1218 GL 89-19	All	E / (3/90)	See Note 17
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	NA	
A-49	Pressurized Thermal Shock	RGs 1.154, 1.99 SECY 82-465 SECY 83-288 SECY 81-687 10 CFR 50.61	PWR	C / (10/86)	See Note 18

Note 1: Water Hammer

As discussed in the NRC Safety Evaluation Report (SER), Appendix C (November 1983), the NRC determined that the probability of a significantly damaging waterhammer event was very low and as such, that the existing plant design and operational controls would likely be adequate. The resolution of USI A-1 did not result in any new requirements for SHNPP.

Note 2: Asymmetric LOCA Loads

As documented in the NRC SER Section 3.9.2.4 (November 1983), resolution of USI A-2 for SHNPP is complete.

Note 3: S/G Tube Integrity

By letter dated June 17, 1985, CP&L submitted the SHNPP response to Generic Letter 85-02 detailing proposed plant modifications and procedure changes made in response to USI A-3. The NRC in SER Supplement 3, Section 5.4.2.2 (May 1986) and SER Supplement 4, Section 5.4.2.2 (October 1986) found these changes acceptable. All modifications and procedure changes were completed prior to the issuance of the Operating License (January 1987).

Note 4: ATWS

As documented in NRC Inspection Report 50-400/89-13, dated July 7, 1989, SHNPP has completed the necessary plant modifications, procedure changes, and training required as a result of USI A-9.

Note 5: Reactor Vessel Material Toughness

As documented in the NRC SER, Section 5.3.1.3 (November 1983), all SHNPP reactor vessel beltline materials have predicted Charpy upper shelf energies greater than 50 ft-lbs at end-of-life. As such, there is no required implementation date and USI A-11 is not applicable to SHNPP.

Note 6: Fracture Toughness of S/G and RCP Supports

As documented in the NRC SER, Appendix C (November 1983), the resolution of USI A-12 was completed with the issuance of NUREG 0577, Revision 1. The new requirements apply to new construction permits and preliminary design approvals plants only and as such resolution of USI A-12 resulted in no changes for SHNPP.

Note 7: Systems Interaction

NRC Generic Letter 89-17, dated September 6, 1989, describes the resolution of USI A-17. The Generic Letter does not require any licensee action. As such, no changes are necessary at SHNPP.

Note 8: Qualification of Class 1E Equipment

As documented in NRC SER Supplement 4, Section 3.11 (October 1986), the NRC has determined that SHNPP has demonstrated full compliance with the requirements for environmental qualifications as detailed in 10 CFR 50.49, the relevant parts of GDC 1 and 4, and Sections III, XI, and XVII of Appendix B to 10CFR50, and with the criteria specified in NUREG-0588. As such, resolution of USI A-24 is complete for SHNPP.

Note 9: Reactor Vessel Pressure Transient Protection

By letter dated June 30, 1989, CP&L submitted a SHNPP license amendment request which revises the Technical Specification heatup and cooldown curves in response to Generic Letter 88-11. CP&L is awaiting NRC approval of these changes.

Note 10: RHR Shutdown Requirements

As documented in the NRC SER, Section 5.4.7 (November 1983), SER Supplement 4, Section 5.4.7 (October 1986) and the NRC SER transmitted by letter dated September 29, 1988, resolution of requirements of USI A-31 for SHNPP are complete.

Note 11: Control of Heavy Loads

As documented in the NRC SER Supplement 4, Section 9.1.5 (October 1986), resolution of USI A-36 for SHNPP is complete.

Note 12: Seismic Design Criteria

By letter dated October 3, 1989, CP&L submitted plant specific information concerning the design of above-ground vertical steel tanks which should resolve USI A-40 for SHNPP. The NRC SER has not yet been issued.

Note 13: Containment Sump Performance

As discussed in Generic Letter 89-21, dated October 19, 1989, additional NRC guidance revised as a result of the resolution of USI A-43 is only applicable to future construction permits, preliminary design approvals , final design approvals, standardized designs, and applications for licenses to manufacture. As such, these requirements are not applicable to SHNPP.

Note 14: Station Blackout

As discussed in CP&L letter dated March 3, 1989, SHNPP will implement all necessary changes for the resolution of station blackout requirements within two years of NRC issuance of an SER approving the proposed changes.

Note 15: Shutdown Decay Heat Removal Requirements

As discussed in NRC letter (SECY 88-260) issued September 13, 1989, USI A-45 was resolved without imposing any new licensing requirements other than the requirement to complete an Individual Plant Examination (IPE). By letter dated October 31, 1989, CP&L committed to submit the SHNPP PRA in response to the IPE requirements by August 1992.

Note 16: Seismic Qualification

As documented in NUREG 1211 (February 1987), the requirements of USI A-46 are not applicable to SHNPP.

Note 17: Safety Implications of Control Systems

NRC requirements associated with the resolution of USI A-47 are contained in NRC Generic Letter 89-19 dated September 1989. CP&L is presently evaluating these requirements and will submit it plans for implementation at SHNPP by March 19, 1990.

Note 18: Pressurized Thermal Shock

As documented in NRC SER Supplement 4, Section 5.3.4 (October 1986) SHNPP resolution of USI A-49 is complete. By letter dated September 8, 1989, CP&L submitted a reevaluation of SHNPP PTS Reference Temperatures (Attachment 1 Table 7.3) based on the Cycle 1 reactor vessel coupon results as required by SER Supplement 4.

PLANT Harris DOCKET NO(S). 50-400

PROJECT MANAGER R. Becker TECHNICAL CONTACT A. Serkiz

USI NO. A-1 TITLE Water Hammer

MPA NO. N/A TAC NOS. N/A

ISSUES SUMMARY:

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 feed-water 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.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

The staff reviewed the Harris design with respect to the water Hammer issue for the operating license and issued the results of the evaluation in SE, NUREG-1038, Appendix C, November 1983. In their evaluation, the staff concluded that the probability was small for a serious water hammer and that the plant could operate with the present design until a final resolution was adopted.

The final resolution in SECY 84-119 did not impose any additional items for Harris.

REFERENCES:

Plant name: Harris
A-1

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Letter from Denton to Utilities, "Notice of Issuance and Availability NUREG-0927 Rev. 1, Safety Issue A-1"	8403150310	03/05/84
NUREG-0927 "Evaluation of Water Hammer in Nuclear Power Plants-Technical Findings Relevant to Unresolved Safety Issue A-1"	8306060413	05/31/83
NUREG-0993 Rev. 1 "Regulatory Analysis for for USI A-1, Water Hammer"	8306060418	March 1984
SRP Sections: 3.9.3, 3.9.4, 5.4.6, 5.4.7, 6.3, 9.2.1, 9.2.2, 10.3, and 10.4.7		

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Safety Evaluation Report, NUREG-1038	8312230068	11/83
SECY-84-119, "Resolution of Unresolved Safety A-1, Water Hammer"		3/15/84

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Harris DOCKET NO(S). 50-400

PROJECT MANAGER R. Becker TECHNICAL CONTACT Jai Rajan

USI NO. A-2 TITLE Asymmetric Blowdown Loads in RCS

MPA NO. D-10 TAC NOS.

ISSUES SUMMARY:

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 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 asymmetric blowdown loads issue in PWRs primary coolant main loop piping.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

The staff reviewed the Harris design with respect to asymmetric blowdown loads in the RCS to issue the operating license and issued the results of the evaluation in the SE, NUREG-1038, Appendix C, November 1983. In their evaluation, the staff concluded that the probability was small for asymmetric blowdown loads in the RCS and that the plant could operate with the present design until a final resolution of this issue was adopted.

The final resolution (GL 84-04 and GDC-4) did not impose any additional items for Harris.

REFERENCES:

Plant Name: Harris
A-2

1. **REQUIREMENT DOCUMENTS:**

TITLE	NUDOCS NO.	DATE
Generic Letter "Evaluation of Primary Systems for Asymmetric LOCA Loads"		01/20/78
Task Action Plan A-2, "Asymmetric Blowdown Loads on Reactor Primary Coolant System," NUREG-0371 Task Action Plans for Generic Activities		11/78
"Asymmetric Blowdown Loads on PWR Primary Systems," NUREG-0609 US NRC NRR		01/81
GDC-4, "Environmental and Dynamic Effects Design Basis"		
GL 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops."		

2. **IMPLEMENTATION DOCUMENTS:**

TITLE	NUDOCS NO.	DATE
(1) WCAP 9558, Rev. 2 "Mechanistic Fracture Evaluation of Reactor Coolant Pipe containing a Postulated Circumferential Throughwall crack"		5/81
(2) WCAP 9787 "Tensil and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation"		5/81
(3) Letter from E.P. Rahe (W) to D.G. Eisenhut (NRC "Westinghouse Response to Questions and Comments raised by members of ACRS subcommittee on Metal Components during the Westinghouse Presentation on September 25, 1981."		11/10/81
(4) GDC-4, 10 CFR 50		
(5) Safety Evaluation Report, NUREG-1038	8312230068	11/83

Plant Name: Harris
A-2

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Harris DOCKET NO(S). 50-400

PROJECT MANAGER R. Becker TECHNICAL CONTACT J. Mauck

USI NO. A-9 TITLE ATWS per 10 CFR 50.62

MPA NO. A-20 TAC NOS. 64561

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.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

On July 26, 1984, the Code of Federal Regulations was amended to include 10 CFR 50.62. In response to 50.62, Westinghouse owners group (WOG) submitted conceptual ATWS mitigating system designs in WCAP-10858 which was reviewed and a safety evaluation issued on July 7, 1986 as WCAP-10858-P-A. Subsequently, Revision 1 was issued for this topical. Harris submitted plant specific submittals on January 11, April 12 and May 10, 1988. The Harris ATWS mitigation design was found to be in compliance with the requirements of 50.62 and the SE was issued on July 14, 1988. System installation proceeded and was verified by IR 50-400/89-13 issued 7/07/89. Review of the records could not establish a licensee implementation date. Therefore, by agreement of the licensee and project manager the implementation date was taken as the date of regional verification of 7/7/89 by IR 50-400/89-13.

REFERENCES:

Plant Name: Harris
A-9

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
NUREG-0460, and Supplements, "Anticipated Transients Without Scram for Light Water Reactors"		03/80
Federal Register Notice 49 FR 26045 (10 CFR 50.62)		06/26/84

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
(1) Safety Evaluation Report, NUREG-1038	8312230068	11/83
WCAP-10858-P-A (Safety Evaluation)		7/07/86
Letter CP&L to NRC	8801200120	1/11/88
Letter CP&L to NRC	8804210144	4/12/88
Letter CP&L to NRC	8805190171	5/10/88
Letter NRC to CP&L (Safety Evaluation)	8807190149	7/14/88

VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
IR 50-400/89-13	8907180194	7/07/89

PLANT Harris DOCKET NO(S). 50-400

PROJECT MANAGER R. Becker TECHNICAL CONTACT B. Elliott

USI NO. A-11 TITLE Reactor Vessel Materials Toughness

MPA NO. A-7 TAC NOS. 71500

ISSUES SUMMARY:

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.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

The Staff reviewed the Harris design with respect to Reactor Vessel Materials Toughness issue for the operating license and issued the evaluation in SE, NUREG-1038, Section 5.3.1.3, November 1983. In their evaluation, the staff concluded that because the Charpy upper shelf energy at the end-of-life was predicted to exceed 50 ft-lb, the requirement of Appendix G charpy upper shelf energy was satisfied. Therefore, the staff considers the status of this USI to have no change.

REFERENCES:

Plant Name: Harris
A-11

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
NUREG-0744, Revision 1, "Pressure Vessel Material Fracture Toughness"		10/82
Generic Letter 82-26, "Pressure Vessel Material Fracture Toughness"		11/12/82

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Safety Evaluation Report NUREG-1038	8312230068	11/83

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Harris DOCKET NO(S). 50-400

PROJECT MANAGER R. Becker TECHNICAL CONTACT A. Thadani

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

MPA NO. TAC NOS.

ISSUES SUMMARY:

Generic Letter (GL) 89-18, dated September 6, 1989, was sent to all power reactor licensees and constitutes the resolution of USI A-17. No licensee actions were required by GL 89-18.

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 "Individual Plant Examination (IPE) for Severe Accident Vulnerabilities." Specifically, the staff expected that insights concerning water intrusion and flooding form internal sources, provided in the appendix to NUREG-1174, will be considered in the IPE program. Also considered in this USI's resolution was the expectation that licensees would continue to review information on events at operating nuclear power plants in accordance with the requirements of Item I.C.5 of NUREG-0737.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

Harris was licensed in January 1987. Therefore, the 1972 letter from research was not sent to Carolina Power & Light Company for the Harris Plant.

REFERENCES:

Plant name: Harris
A-17

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Generic Letter 89-18		09/06/89
NUREG-1174 "Evaluation of Systems Interactions in Nuclear Power Plants"		
NUREG-1229 "Regulatory Analysis for Resolution of USI A-17"		
NUREG/CR-3922 "Survey and Evaluation of System Interaction Events and Sources"		January 1985
NUREG/CR-4261 "Assessment of System Interaction Experience in Nuclear Power Plants"		June 1986
NUREG/CR-4470 "Survey and Evaluation of Vital Instrumentation and Control Power Supply Events"		August 1986

IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

3. VERIFICATION DOCUMENTS:

TITLE	NUDOC NO.	DATE

PLANT Harris DOCKET NO(S). 50-400

PROJECT MANAGER R. Becker TECHNICAL CONTACT P. Shemanski

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

MPA NO. TAC NOS.

ISSUES SUMMARY:

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.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

The staff reviewed the licensee's equipment qualification program in SERs NUREG-1038, Section 3.11, November 1983 and Supplement 4, October 1986. The staff accepted the licensee's program as being in full compliance in NUREG-1038, Supplement 4, 1986. The plant was licensed for full power operation in January 1987.

REFERENCES:

Plant Name: Harris
A-24

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
DOR "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors"		
NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment"		12/79
Commission Memorandum and Order, CLI-80-21, on DOR Guidelines and NUREG-0588		05/23/80
NUREG-0588, Revision 1		07/81
10 CFR 50.49 (48 FR 2730-2733)		01/21/83
Standard and Review Plan 3.11, Environmental Qualification of Mechanical and Electrical Equipment		07/81

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Forwards Specific date on EQ	8011070436	11/3/80
Forwards responses to draft SER	8306270100	6/22/83
Forwards EQ master list	8408060192	7/27/84
Forwards site visit trip report	8605210046	8/27/85
Notifies of qualification status	8610030438	5/15/86
Safety Evaluation Report, NUREG-1038, Supplement 4	8611050030	10/86

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Harris DOCKET NO(S). 50-400

PROJECT MANAGER R. Becker TECHNICAL CONTACT Chu Liang

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

MPA NO. B-4 TAC NOS. 71500

ISSUES SUMMARY:

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.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

The staff reviewed the issue originally in SER,section 5.2.2 NUREG-1038 November, 1983. Licensee submitted additional information closing SER open items 4/23/86 which established the LTOP set points.

REFERENCES:

Plant Name: Harris
A-26

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
NUREG-0224 - "Reactor Vessel Pressure Transient Protection for PWRs."		9/78
NRC Letters to Licensees Informing Licensees of Staff Concerns Regarding Overpressure Low-Temperature Conditions in PWRs		August 1976
Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations"		7/12/88
Standard Review Plan Section 5.2		

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
(1) Safety Evaluation Report, NURE-1038	8312230068	11/83
(2) Letter CP&L to NRC	8604280091	4/23/86

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Harris DOCKET NO(S). 50-400

PROJECT MANAGER R. Becker TECHNICAL CONTACT R. Jones

USI NO. A-31 TITLE RHR Shutdown Requirements

MPA NO. TAC NOS.

ISSUES SUMMARY:

This USI was resolved in may 1978 and affected PWRs and BWRs. 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.

The USI was resolved with the issuance of Standard Review Plan (SRP) Section 5.4.7. The SRP stated (Branch Technical Position RSB 5-1) that, for purposed of implementation, plants would be divided into three classes: Class 1 would require full compliance with the Position for CP or PDA applications which were docketed on or after January 1, 1978. Class 2 required a partial implementation of the Position for all plants for which CP or PDA applications were docketed before January 1, 1978, and for which an 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 of the Position was bases on the combined I&E and DOR review of related plant features. In general, the outcome of these evaluations were that only plants receiving OL after January 1, 1979 were affected by this USI resolution.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

The staff reviewed the Harris design with respect to RHR shutdown requirements and issued the results in the licensing Safety Evaluation, NUREG-1038, 1983, and Supplement 4, 1986 Section 5.4.7. NUREG-1038, Supplement 4 required a comparison between Natural Circulation in Diablo Canyon, Unit 1, and Harris in lieu of performing the natural circulation test in Harris. A review and acceptance of the comparison was transmitted to the licensee by letter dated September 29, 1983.

REFERENCES:

Plant Name: Harris
A-31

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
NUREG-0800 "Standard Review Plan," SRP Section 5.4.7		5/78
NUREG-0606 "Unresolved Safety Issues Summary"		
Regulatory Guide 1.139, "Guidance for Residual Heat Removal"		
Regulatory Guide 1.113		

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Safety Evaluation Report NUREG-1038, Safety	8312230068	11/83
Evaluation Report NUREG-1038, Supplement 4	8611050030	10/86
Letter NRC to CP&L		9/29/88

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Harris DOCKET NO(S). 50-400

PROJECT MANAGER R. Becker TECHNICAL CONTACT J. Wermiel

USI NO. A-36 TITLE Control of Heavy Loads, Phases I & II

MPA NO. C-10, C-15 TAC NOS.

ISSUES SUMMARY:

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.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

The staff's review of the licensee's response to this concern is documented in the operating licensee SE, NUREG-1038, Section 9.1, Supplement 4, October 1986.

REFERENCES:

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Letter, Darrell G. Eisenhut, NRC, to all licensees, applicants for OLs and holders of CPs transmitting NUREG-0612 and staff positions		12/22/80
Generic Letter 85-11, Hugh L. Thompson, NRC, to all licensees for Operating Reactors, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants' NUREG-0612"		06/28/85

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
(1) Safety Evaluation Report NUREG-1038	8312230068	11/83
(2) Safety Evaluation Report, NUREG-1038, Supplement 4	8611050030	10/86

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Harris DOCKET NO(S). 50-400

PROJECT MANAGER R. Becker TECHNICAL CONTACT H. Ashar

USI NO. A-40 TITLE Seismic Design Criteria

MPA NO. — TAC NOS. 73096

ISSUES SUMMARY:

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 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.

Plant Name: Harris
A-40

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

The staff concluded in the operating license SER, NUREG-1038, Supplement 1, November 1983 that the seismic design of Harris was acceptable until the final resolution of A-40. Harris was one of the licensees required to respond about design of vertical steel tanks utilizing SQUG recommended methodology. RAI letter was sent to the licensees on June 1, 1989. The licensee responded on October 3, 1989. Response is presently (11/3/89) under review by the Structural and Geosciences Branch of NRR.

REFERENCES:

Plant Name: Harris
A-40

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Regulatory Analysis for USI A-40	NUREG-1233	Sept. 1989
Recommendations for Resolution of Public Comments on USI A-40	NUREG/CR-5347	June 1989
Standard Review Plan Sections 2.5.2, 3.7.1, 3.7.2, 3.7.3 (Revision 2)	NUREG-0800	To be issued
Response of Seismic Category I Tanks to Earthquake Excitation	NUREG/CR-4776	Feb. 1987
Engineering Characterization of Ground Motion, Vols. 3,4,5	NUREG/CR-3805	Feb.-Aug. 1986
Proceedings of the Workshop on Soil-Structure Interaction	NUREG/CR-0054	June 1986
Value Impact Assessment for Seismic Design Criteria	NUREG/CR-3480	Aug. 1984
Seismic Hazard Analysis Application of Methodology, Results and Sensitivity Studies, Vol. 4	NUREG/CR-1582	Oct. 1981
Recommended Revision to Nuclear Regulatory Commission Seismic Design Criteria	NUREG/CR-1161	May 1980
Power Spectral Density Functions Compatible with NRC R.G. 1.60 Response Spectra	NUREG/CR-3509	June 1988

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Request for Information Letters to Owner's of Callaway 1&2, Wolf Creek 1, Shearon Harris 1, Watts Bar 1&2	8906200356	6/1/89
Letter CP&L to NRC	8910110175	10/3/89

Plant Name: Harris
A-40

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Harris DOCKET NO(S). 50-400

PROJECT MANAGER R. Becker TECHNICAL CONTACT P. Gill

USI NO. A-44 TITLE Station Blackout

MPA NO. A-22 TAC NOS. 68552/40577

ISSUES SUMMARY:

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.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

The rule was issued July 21, 1988. The licensee was requested to submit a coping analysis. Coping analysis submitted March 3, 1989. Staff review expected to be complete by March 31, 1991. No hardware changes were required by the licensee to meet the rule.

REFERENCES:

Plant Name: Harris
A-44

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
10 CFR 50.63, "Loss of All Alternating Current Power"		06/21/88
Regulatory Guide 1.155, "Station Blackout"		08/88

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Letter CP&L to NRC Response to Station Blackout Rule (TAC 68552)	8903140107	3/3/89

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Harris DOCKET NO(S). 50-400

PROJECT MANAGER R. Becker TECHNICAL CONTACT P. Y. Chen

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

MPA NO. B-105 TAC NOS. 73096

ISSUES SUMMARY:

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.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

Refer to A-40 for additional and plant specific detail.

REFERENCES:

Plant Name
A-46

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Generic Letter 87-02, "Verification of Seismic Adequacy of Mechanical and Electric Equipment in Operating Reactors"		02/19/87
NUREG-1211, "Regulatory Analysis for Resolution of Unresolved Safety Issues A-46..."		02/87
NUREG-1030, "Seismic Qualification of Equipment in Operating Plants, Unresolved Safety Issue A-46"		02/87
Letter attached with "Generic Safety Evaluation Report on SQUG GIP, Revision 0," from L. Shao (NRC) to Neil Smith (SQUG)		07/29/88

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
"Generic Implementation Procedure (GIP for Seismic Verification of Nuclear Plant Equipment," Revision 0		06/88
"Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Revision I		12/88

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Harris DOCKET NO(S). 50-400

PROJECT MANAGER R. Becker TECHNICAL CONTACT J. Mauck

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

MPA NO. ________ TAC NOS. 74948

REQUIREMENTS SUMMARY:

USI A-47 is being closed out with issuance of Generic Letter (GL) 88-19 which was issued September 20, 1989. The GL 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."

The GL provides requirements that licensees provide NRC with their schedule and commitments within 180 days of the GL 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 GL.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

Licensee is still evaluating the Generic Letter.
Response is not expected until March 1990.

REFERENCES:

Plant name: Harris
A-47

1. REQUIREMENT DOCUMENTS

TITLE	NUDOCS NO.	DATE
Generic Letter 89-19 "Request for Action Related to Resolution of USI A-47"		09/20/89
NUREG-1217 "Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plants"		June 1989
NUREG-1218 "Regulatory Analysis for Resolution of USI A-47"		July 1989

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Harris DOCKET NO(S). 50-400

PROJECT MANAGER R. Becker TECHNICAL CONTACT B. Elliott

USI NO. A-49 TITLE Pressurized Thermal Shock

MPA NO. TAC NOS.

ISSUES SUMMARY:

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.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

The licensee first evaluated USI A-46 in the operating license SE, NUREG-1038, Appendix C, November 1983 and concluded that the Harris design was adequate for operation until the final resolution was adopted. The limiting materials in the reactor vessel of Harris, Unit 1 are B-4197-2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," dated July 12, 1988 requested licensees to submit an evaluation of using the new RG 1.99, Revision 2 methodology and a schedule for any additional follow-up actions. The licensee responded to GL 88-11 by letter dated January 6, 1989 and submitted a request for license amendment changes based on the RG 1.99, Rev. 2 methodology on June 30, 1989 which is currently (11/3/89) under review. The new methodology would decrease RT_{NDT} for B-4197-2 by 4°F to 86°F.

REFERENCES:

Plant Name: Harris
A-49

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Requirements"		7/85
Reg. Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for PWRs"		1/89
SECY 82-465, "Pressurized Thermal Shock"		11/23/82
SECY 83-288, "Proposed Pressurized Thermal Shock Rule"		07/15/83
Regulatory Guide 1.154 "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors"		02/87
Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations"		7/12/88

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Letter CP&L to NRC Generic Letter 88-11 Response	8901100421	1/6/89
Amendment Request	8907070180	6/30/89

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

ENCLOSURE 3

CATEGORIES FOR USI STATUS

(1) Where an item is not applicable to the facility, "NA" is entered in the status column.

(2) Where an item is applicable to the facility, but no changes were necessary, "NC" is entered in the status column. Also, "NC" is entered if the USI was implemented prior to licensing, as no changes were necessary after issuance of an OL. No implementation dates are entered for items that are "NC."

(3) Where an item is applicable to the facility and changes are complete, "C" is entered in the status column and the date implementation was completed is entered.

(4) Where an item is applicable to the facility and is not fully implemented, "I" is entered in the status column and the projected implementation date is entered.

(5) Where a USI resolution was recently issued and the licensee's evaluating their response, "E" is entered in the status column and the projected response date is entered, if known.

LISTING OF INCOMPLETE USI DATA
FOR INPUT FROM PROJECT MANAGERS

ISSUE NUMBER	ISSUE DESCRIPTIVE NAME	IMPLEMENT DATE	IMPLEMENT STATUS	LICENSEE COMMENT	STAFF COMMENT
** PLANT NAME: HARRIS 1					
A-01	WATER HAMMER	/ /	NC		SER 8312230068
A-02	ASYMMETRIC BLOWDOWN LOADS ON REACTOR PRIMARY COOLANT SYSTEMS	/ /	NC		
A-03	WESTINGHOUSE STEAM GENERATOR TUBE INTEGRITY	/ /	NC		INFO ONLY
A-04	CE STEAM GENERATOR TUBE INTEGRITY	/ /	N/A		CE PLANTS ONLY
A-05	B&W STEAM GENERATOR TUBE INTEGRITY	/ /	N/A		B&W PLANTS ONLY
A-06	MARK I SHORT-TERM PROGRAM	/ /	N/A		MK I BWR ONLY
A-07	MARK I LONG-TERM PROGRAM	/ /	N/A		MK I BWR ONLY
A-08	MARK II CONTAINMENT POOL DYNAMIC LOADS - LONG-TERM PROGRAM	/ /	N/A		MK II BWR ONLY
A-09	ATWS	07/07/89	C		INSPECTION DATE
A-10	BWR FEEDWATER NOZZLE CRACKING	/ /	N/A		BWR ONLY
A-11	REACTOR VESSEL MATERIALS TOUGHNESS	/ /	NC		
A-12	FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS	/ /	N/A		CP AFTER 83 ONLY
A-17	SYSTEMS INTERACTION	/ /	NC	IPE	NO REQUIREMENTS
A-24	QUALIFICATION OF CLASS 1E SAFETY-RELATED EQUIPMENT	/ /	NC		
A-26	REACTOR VESSEL PRESSURE TRANSIENT PROTECTION	/ /	NC		LTOPS @ OL
A-31	RHR SHUTDOWN REQUIREMENTS	/ /	NC		LICENSING SER
A-36	CONTROL OF HEAVY LOADS NEAR SPENT FUEL	/ /	NC		GL-85-11 ENDED
A-39	DETERMINATION OF SAFETY RELIEF VALVE POOL DYNAMIC LOADS AND TEMPERATURE LIMITS	/ /	N/A		BWR ONLY
A-40	SEISMIC DESIGN CRITERIA - SHORT-TERM PROGRAM	/ /	NC		OL SER
A-42	PIPE CRACKS IN BOILING WATER REACTORS	/ /	N/A		BWR ONLY
A-43	CONTAINMENT EMERGENCY SUMP PERFORMANCE	/ /	NC		INFO ONLY
A-44	STATION BLACKOUT	03/31/93	I		SER 3/31/91
A-45	SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS	/ /	NC	IPE	SUBSUMED BY SEVERE ACC
A-46	SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS	/ /	N/A		OLD PLANTS ONLY
A-47	SAFETY IMPLICATIONS OF CONTROL SYSTEMS	03/19/90	E		NEW REQUIREMENTS
A-48	HYDROGEN CONTROL MEASURES AND EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT	/ /	N/A		N/A DRY CONTAIN
A-49	PRESSURIZED THERMAL SHOCK	/ /	NC		