Indiana Michigan
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INDIANA
MICHIGAN
POWER

AEP:NRC:1108A

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
NRC GENERIC LETTER 89-21: ADDITIONAL INFORMATION

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Attn: T. E. Murley

December 7, 1989

Dear Dr. Murley:

This letter and its attachment provide additional information and revised pages of attachment to our previous letter AEP:NRC:1108 dated November 29, 1989. The revisions are being made as a result of our conversation with the NRR Project Manager.

This document has been prepared following Corporate procedures that incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Sincerely,

M. P. Alexich
Vice President

MPA/eh

Attachment

cc: D. H. Williams, Jr.
A. A. Blind - Bridgman
R. C. Callen
G. Charnoff
NFEM Section Chief
A. B. Davis - Region III
NRC Resident Inspector - Bridgman

8912140402 5pp.

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	NA AEP:NRC:0398 and 0678 Series of letters	Based on enclosure 2 to GL 89-21
A-2/ MPA D-10	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	NUREG-0609 GL 84-04, GDC-4	PWR	C AEP:NRC:0137D Dated: Sept. 10, 1984	NRC SER dated Nov. 22, 1985 Amendment No. 76 to Operating License No. DPR-74.
A-3	Westinghouse Steam Generator Tube Integrity	NUREG-0844 SECY 86-97 SECY 88-272 GL 85-02 (No requirements)	W-PWR	C AEP:NRC:0936 dt. June 21, 1985	Cook Unit 2 steam generators were replaced; for Unit 1 additional information is being provided to NRC under IE Bulletin 88-02
A-4	CE Steam Generator Tube Integrity	NUREG-0844, SECY 86-97 SECY 88-272 GL 85-02 (No requirements)	CE-PWR	-NA-	None
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-	None
E ~~A~~-6	Mark I Containment Short-Term Program	NUREG-0408	Mark I-BWR	-NA-	None

* 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	-NA-	None
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-	None
A-9	Anticipated Transients Without Scram	NUREG-0460, Vol. 4 10 CFR 50.62	All	C AEP:NRC:0838 Series of letters	
A-10/ MPA B-25	BWR Feedwater Nozzle Cracking	NUREG-0619 Letter from DG Eisenhut dated 11/13/80 GL 81-11	BWR	-NA-	None
A-11	Reactor Vessel Material Toughness	NUREG-0744, Rev. 1 10 CFR 50.60/ 82-26	All	-NC-	
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	NUREG-0577, Rev. 1 SRP Revision 5.3.4	PWR	-NC-	
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 in cover letter, Item 1
A-24/ MPA B-[illegible]	[illegible]ification of Class [illegible]ty-Related Equipment	NUREG-0588, Rev. 1 SRP 3.11 10 CFR 50.49 GL 82-09, GL 84-24 GL 85-15	All	-C-	Our letters: AEP:NRC:0356 Series AEP:NRC:0578 Series AEP:NRC:0775 Series

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	C AEP:NRC:0894K dt. - Dec. 5, 1988 AEP:NRC:0894L dt. - Oct. 25, 1989 NRR Inspection Report 50-315, 316/86-34 dt. 11/4/86	See cover letter, Item 2 T/S specification changes are in progress
A-31	Residual Heat Removal Shutdown Requirements	NUREG-0606 RG 1.113, RG 1.139 SRP 5.4.7	All OLs After 01/79.	-NA-	
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 AEP:NRC:0514 Series of letters NRC SER (Phase I) dt. Sept. 20, 1983	See cover letter, Item 3
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-	None
A-40	Seismic Design Criteria	SRP Revisions, NURLG/ 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	-NC-	See cover letter, Item 4
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-	None

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	C	No NRC submittal is required
A-44	Station Blackout	RG 1.155 NUREG-1032 NUREG-1109 10 CFR 50.63	All	I, our letter AEP:NRC:0537D dt. April 14, 1989	Awaiting NRC review
A-45	Shutdown Decay Heat Removal Requirements	SECY 88-260 NUREG-1289 NUREG/CR-5230 SECY 88-260 (No requirements)	All	C, AEP:NRC:1082 dt. Oct. 24, 1989	This has been incorporated into IPE program under GL 88-20
A-46	Seismic Qualification of Equipment in Operating Plants	NUREG-1030 NUREG-1211/ GL 87-02, GL 87-03	All	AEP:NRC:1040 I	See cover letter, Item 5
A-47	Safety Implication of Control Systems	NUREG-1217, NUREG 1218 GL 89-19	All	E	Under review and evaluation by AEPSC
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	I AEP:NRC:0476 and AEP:NRC:0500 Series	Remaining open items will be addressed in the IPE Program by AEPSC
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	C, AEP:NRC:0561A dt. January 22, 1986	NRC SER dated March 27, 1987

ENCLOSURE 2

USI STATUS SUMMARY

PLANT D.C. Cook, Units 1 and 2 DOCKET NO(S). 50-315 AND 50-316

PROJECT MANAGER J. G. Giitter TECHNICAL CONTACT Jai Rajan

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

MPA NO. D-10 TAC NOS. 08477 and 08478

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):

On February 1, 1984, the staff issued Generic Letter 84-04--a safety evaluation of the Westinghouse topical reports dealing with elimination of postulated pipe breaks in PWR primary loops. The safety evaluation concluded that an acceptable basis had been provided so that asymmetric blowdown loads

resulting form double ended pipe breaks in main coolant loop piping need not be considered as a design basis for the Westinghouse Owner's Group plants (including D. C. Cook) provided that two conditions are met. The first condition, which did not apply to Cook, involved verification of bending moments at two other nuclear power plants. The second condition was that leakage detection systems exist to detect postulated flaws utilizing guidance from Regulatory Guide 1.45 (with the exception of seismic equipment qualification for the airborne particulate radiation monitor). At least one leakage detection system sensitive enough to detect a 1 gpm leak within 4 hours was required to be operable.

The licensee responded in letter dated September 10, 1984, indicating that the leak detection systems at Cook are consistent with the requirements of Generic Letter 84-04. In this same letter the licensee proposed to remove license condition C.3(a) to Operating License No. DPR-74 for Unit No. 2, which requires that an analytical evaluation be made of the effects of certain postulated break loads on the reactor coolant system and internals. This license condition was removed by Amendment No. 76 dated November 22, 1985.

REFERENCES:

Cook Units 1 and 2
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

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TITLE	NUDOCS NO.	DATE
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."		02/01/84

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
(1) LTR from Alexich (AEP) to Denton	8409130354	9/10/84

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT D. C. Cook Units 1 and 2 DOCKET NO(S). 50-315 and 50-316

PROJECT MANAGER J. G. Giitter TECHNICAL CONTACT J. Mauck

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

MPA NO. A-20 TAC NOS. 59082 and 59083

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):

In response to paragraph 50.62 (C)(1), the Westinghouse Owners Group (WOG) developed a set of conceptual ATWS mitigating system actuation circuitry (AMSAC) designs for Westinghouse PWRs. These designs were described in Westinghouse Topical Report, WCAP-10858, "AMSAC Generic Design Package". The staff reviewed WCAP-10858 and issued a safety evaluation on July 7, 1986 indicating that the generic designs presented in the WCAP adequately meet the requirements of 10 CFR 50.62. A revision to the WCAP involving a change in the AMSAC permissive signal was also found to be acceptable by the staff.

In a letter dated November 7, 1986 the licensee transmitted preliminary information on the detailed design of the AMSAC proposed for installation at Cook. The licensee provided additional information related to the AMSAC design in letters dated June 25, 1987, October 28, 1987, December 18, 1987, March 31, 1988, and May 2, 1988. The staff completed a safety evaluation of the Cook AMSAC design on July 1, 1988. The safety evaluation concluded that the AMSAC design proposed for D. C. Cook is acceptable provided that electrical isolation devices are successfully qualified. The safety evaluation was transmitted to the licensee in a letter dated April 14, 1989. AMSAC was implemented for Unit 1 during the 1989 refueling outage which ended on July 8, 1989. AMSAC was implemented for Unit 2 during the steam generator replacement outage which ended in February 1989. NRC Inspection Report 89-032 concluded that the installation and testing of AMSAC at D. C. Cook is acceptable.

REFERENCES:

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
Stang (NRC) to Alexich	8904210297	4/14/89

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Cook Inspection Report # 89-032	TBD	11/13-14/89

PLANT D.C. Cook, Units 1 and 2 DOCKET NO(S). 50-315 and 50-316

PROJECT MANAGER J. G. Giitter TECHNICAL CONTACT B. Elliott

USI NO. A-11 TITLE Reactor Vessel Materials Toughness

MPA NO. A-07 TAC NOS.

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):

In their response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its impact on Plant Operations", dated December 5, 1988, the licensee indicated that adequate toughness (i.e., Charpy USE less than 50 ft-lb) for the surveillance capsule specimens of controlling material existed through 32 EFPY (which corresponds to the design life of the plant).

REFERENCES:

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
Alexich (AEP) to NRC	8812090036	12/5/88

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT D. C. Cook, Units 1 and 2 DOCKET NO(S). 50-315 and 50-316

PROJECT MANAGER J. G. Giitter TECHNICAL CONTACT R. Johnson (RES)

USI NO. A-12 TITLE Potential of Low Fracture Toughness and Lamellar Tearing in PWR SG and RCP Supports

MPA NO. A-07 TAC NOS.

ISSUES SUMMARY:

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.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

The licensee responded to the NRC's request for information concerning the fracture toughness of the steam generator and reactor coolant supports at D. C. Cook Units 1 and 2 in a letter dated November 23, 1977. This information, initially reviewed by Sandia, was independently reviewed by Franklin Research Center. Franklin reviewed the information per the criteria presented in NUREG-0577 and concluded the supports possess adequate fracture toughness. The staff concurred with the this conclusion in a safety evaluation that was transmitted to the licensee on December 10, 1980.

REFERENCES:

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
NUREG-0577, Rev. 1, "Potential of Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports"		10/83

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
NRC to AEP	8101080364	12/10/80

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT D. C. Cook, Units 1 and 2 DOCKET NO(S). 50-315 and 50-316

PROJECT MANAGER J. G. Giitter TECHNICAL CONTACT D. Thatcher

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. 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).

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

AEP is currently reviewing Generic Letter 89-18. The licensee stated in their November 29, 1989 response to Generic Letter 89-21 that systems interaction (e.g., internal flooding) will be addressed to some extent in their IPE program. A brief description of the IPE program was provided to the NRC in a letter dated October 24, 1989.

A search of appropriate documents (i.e., Safety Evaluation and Supplements, FSAR, etc) for a potential licensee response or staff consideration of the September 1972 letter provided negative results. Similarly, a preliminary search by the licensee did not yield a record of the September 1972 letter.

REFERENCES:

Cook Units 1 and 2
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"		May 1989
NUREG-1229 "Regulatory Analysis for Resolution of USI A-17"		August 1989
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
NRC Letters to Licensees Informing Licensees of Staff Concerns Regarding Potential Failure of Non-Category I Equipment		9/72

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

3. VERIFICATION DOCUMENTS:

TITLE	NUDOC NO.	DATE

PLANT D. C. Cook, Units 1 and 2 DOCKET NO(S). 50-315 and 50-316

PROJECT MANAGER J G. Giitter 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 issued orders dated August 29, 1980 (amended in September 1980) and October 24, 1980 to all licensees. The August order required that the licensees provide a report, by November 1, 1980, documenting the qualification of safety-related electrical equipment. The October order required the establishment of a central file location for the maintenance of all equipment qualification records. The central file was mandated to be established by December 1, 1980. The staff subsequently issued a safety evaluation on environmental qualification of safety-related electrical equipment to the licensee on May 26, 1981. The SER directed the licensee to "either provide

documentation of the missing qualification information which demonstrates that safety-related equipment meets the DOR Guidelines or NUREG-0588 requirements or commit to a corrective action." The information submitted by the licensee in response to the May 26, 1981 safety evaluation was evaluated for the staff by Franklin Research Center (FRC). A safety evaluation transmitting the TER was sent to the licensee on January 17, 1983. On February 22, 1983 the final rule (10 CFR 50.49) on environmental qualification of electrical equipment became effective.

On September 13, 1983, a meeting was held with the licensee to discuss resolution of environmental qualification deficiencies identified in the January 17, 1983 safety evaluation. Subsequent submittals dated January 17, June 12, October 18 and December 10, 1984 documented the licensee's proposed method of resolving environmental qualification deficiencies and complying with 10 CFR 50.49. On January 11, 1985, the staff issued a safety evaluation that concluded that the licensee's equipment qualification program is in compliance with 10 CFR 50.49 based on discussions during the September 13, 1983 meeting and review of the subsequent 1984 submittals.

In a letter dated January 25, 1985 the licensee stated under oath and affirmation : "There is currently in place (at D. C. Cook) an Environmental Qualification (EQ) Program which is establishing the qualification requirements for the applicable electrical equipment. We believe this EQ program satisfies the requirements of 10 CFR 50.49 as we understand them, within the constraints currently approved by the NRC staff's Safety Evaluation (SE) dated January 11, 1985." This letter went on to say that a request for an extension of the environmental qualification deadline was not anticipated.

However, in a letter dated June 28, 1985 the licensee requested an extension of the deadline to allow for the relocation of cables for the steam generator narrow range level differential pressure transmitters that were discovered to be routed below the maximum containment flood elevation. The licensee committed to rerouting the cables for Unit 1 during the current outage (7/85) but requested an extension for Unit 2 until the next scheduled outage. In a letter dated September 11, 1985 the staff informed the licensee that the request for an extension would not be considered. In a letter dated September 30, 1985 the licensee appealed to the Commission to consider an extension for Unit 2. In a letter dated November 14, 1985, the Commission granted an extension until the next scheduled outage (no later than February 28, 1986). The cables for Unit 2 were rerouted during this outage (3/86).

REFERENCES:

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
Varga to Dolan (AEP)	8501290099	1/11/85
Alexich (AEP) to Denton	8501310460	1/25/85
Alexich (AEP) to Palladino	8510070170	9/30/85
Chilk to Dolan (AEP)	8511260216	11/14/85

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT D. C. Cook, Units 1 and 2 DOCKET NO(S). 50-315 and 50-316

PROJECT MANAGER J. G.Giitter TECHNICAL CONTACT Chu Liang

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

MPA NO. B-04 TAC NOS. 06805 and 06806

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 (LTOP) setpoints.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

By letter dated September 9, 1976 the licensee submitted to the NRC their initial response to the staff's August 13, 1976 request for a schedule and information to resolve the potential for overpressurizing the reactor coolant system. A plant specific analysis in support of a reactor vessel overpressure mitigation system for Cook Unit 1 was submitted by the licensee in a letter dated August 4, 1977. The staff concluded in a safety evaluation dated March 11, 1982, that the administrative controls and design modifications proposed by the licensee for Cook Unit 1 meets the criteria proposed by the NRC and is acceptable as a long term solution to the problem of overpressure transients.

The overpressure mitigation system (OMS) for Unit 2 was addressed in Supplement 7 to the D. C. Cook Unit 2 Safety Evaluation Report issued in

December of 1977. The Supplement to the SER found that the OMS for Unit 2 was acceptable subject to the installation of equipment before twelve effective full power months of operation. The licensee completed the installation within this time frame. Technical Specifications covering the OMS for Unit 1 and Unit 2 were incorporated by Amendment 53 and Amendment 39, respectively.

On October 25, 1989 the licensee submitted revised pressure-temperature curves for Unit 2 based on recent surveillance capsule results and Regulatory Guide 1.99, Rev. 2 methodology. Changes to LTOP setpoints were not necessary. After analysis of surveillance capsule U is completed the licensee will submit pressure-temperature curves for Unit 1.

REFERENCES:

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
AEP to NRC		8/4/77
D. C. Cook Unit 2 SER, Supplement 7		12/77

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
NRC to AEP	8204220602	3/11/82

PLANT D. C. Cook, Units 1 and 2 DOCKET NO(S). 50-315 and 50-316

PROJECT MANAGER J. G. Giitter TECHNICAL CONTACT J. Wermiel

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

MPA NO. C-10, C-15 TAC NOS. 07980, 07981, 52217 and 52218

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):

By letter dated December 22, 1980, the staff requested the licensee to review their provisions for handling and control of heavy loads at Cook Units 1 and 2 to determine the extent to which the guidelines of NUREG-0612 are presently satisfied. The staff and its consultant, Franklin Research Center, reviewed the licensee's submittals and issued a safety evaluation on September 20, 1983. The safety evaluation concluded that the guidelines in NUREG-0612, Sections 5.1.1 and 5.3 have been satisfied and, therefore, that Phase I for D. C. Cook Units 1 and 2 is acceptable. The licensee notified the staff by letter dated April 10, 1986 that modification of the drum pinion and gear of the main hoist to meet CMAA-70 (1975) standards had been completed in accordance with a previous commitment.

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
Varga to Dolan (AEP)	8310070408	9/20/83
Alexich (AEP) to Denton	8604160065	4/10/86

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT D. C. Cook, Units 1 and 2 DOCKET NO(S). 50-315 and 50-316

PROJECT MANAGER J. G. Giitter TECHNICAL CONTACT A. Serkiz

USI NO. A-44 TITLE Station Blackout

MPA NO. A-22 TAC NOS. 68532 and 68533

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 licensee responded to the rule in a letter dated April 14, 1989. In their response the licensee stated that the coping duration categories (e.g., AC and

Emergency AC power classification) proposed for Cook are consistent with the guidance of NUMARC 87-00. The submittal also identified procedures that would be modified, as necessary, to meet the guidelines in NUMARC 87-00: These include procedures that identify loads that can be stripped from the batteries to provide a four hour capacity and procedures necessary to ensure backup methods exist for air operated valves needed for decay heat removal. The safety evaluation for D. C. Cook Units 1 and 2 is scheduled to be completed during the second quarter of 1991.

REFERENCES:

Cook Units 1 and 2
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
Alexich (AEP) to NRC	8904260177	4/14/89

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT D C. Cook, Units 1 and 2 DOCKET NO(S). 50-315 and 50-316

PROJECT MANAGER J. G. Giitter TECHNICAL CONTACT P. Y. Chen

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

MPA NO. B-105 TAC NOS.

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):

For All Plants:
The Generic Implementation Procedure (GIP), Revision 0, was submitted by SQUG (of which AEP is a member) on June 3, 1988. The staff issued a Generic Safety Evaluation (SE) on July 29, 1988 endorsing much of the GIP but with about 70 open items to be resolved. After a series of meetings, SQUG submitted Revision I to the GIP on December 23, 1988. Supplemental information was

REFERENCES:

Cook Units 1 and 2
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"		2/19/87
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
Alexich (AEP) to NRC	8810110222	10/3/88

submitted by SQUG on March 17, 1989. The staff has prepared a supplemental SE for GIP, Revision I and has submitted it to CRGR for review. The target date for issuance of the supplemental SE is November 1989. An additional supplement is scheduled for June 1990 and overall closeout of implementation projected for 1993.

Plant Specific
The licensee responded to the NRC request for a plant-specific seismic verification plan for Cook by letter dated October 3, 1988. In their letter the licensee stated that plant walkdown at Cook is expected to be completed by December 1992--provided that there are no major changes to the work scope and that the final SE (with no open items) is issued on schedule.

REFERENCES:

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 D. C. Cook, Units 1 and 2 DOCKET NO(S). 50-315 and 50-316

PROJECT MANAGER J. G. Giitter TECHNICAL CONTACT J. Kudrick

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

MPA NO. .... TAC NOS. 49519 and 49520

ISSUES SUMMARY:

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.

PLANT D. C. Cook, Units 1 and 2 DOCKET NO(S). 50-315 and 50-316

PROJECT MANAGER J. G. Giitter TECHNICAL CONTACT J. Mauck

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

MPA NO. ____ TAC NOS. 79430 and 79431

ISSUES SUMMARY:

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.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

The licensee is currently evaluating this USI.

REQUIREMENTS SUMMARY (CONT.):

The NRC staff has concluded that USI-A-48 is resolved as stated in SECY 89-122. If interested, the report should be consulted for further details regarding the relationship of A-48 to other ongoing hydrogen activities.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

NUREG-1370, Resolution of Unresolved Safety Issue A-48, concluded, "an adequate basis exists for hydrogen control measures for degraded core accidents and that no new regulatory guidance for such accidents is necessary. Therefore, the staff concludes that USI A-48 is resolved". However, NUREG-1370 also notes that the staff has required utility owners of ice-condenser containments to perform additional analyses to demonstrate equipment survivability for a broad spectrum of degraded core accidents and that these efforts (including staff review) will be completed in 1990. Several additional hydrogen control issues have not been resolved for D. C. Cook. The licensee currently plans to address these issues as part of their Individual Plant Evaluation effort.

REFERENCES:

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
10 CFR 50.44, Standards for Combustible Gas System in Light-Water-Cooled Power Reactors		12/81
SECY-89-122, Resolution of USI A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment"		04/19/89

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT D. C. Cook, Units 1 and 2 DOCKET NO(S). 50-315 and 50-316

PROJECT MANAGER J. G. Giitter TECHNICAL CONTACT B. Elliott

USI NO. A-49 TITLE Pressurized Thermal Shock

MPA NO. A-21 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):

By letters dated January 22, 1986 and February 27, 1987, the licensee submitted information on the reactor vessel material properties and the fast neutron fluence in the reactor vessel beltline region for Cook Units 1 and 2. The staff reviewed this information and issued a safety evaluation on March 27, 1987 that concluded that the Reference Temperature for PTS (RT-PTS) for both units is well within the 10 CFR.61 screening criteria through the end of the current operating license. The SER also stated that the licensee should submit a reevaluation of RT-PTS as required by 10 CFR 50.61 whenever core loadings, surveillance measurements, or other information indicate a significant change in projected values.

REFERENCES:

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
Alexich (AEP) to NRC	8601280125	1/22/86
Youngblood to Dolan (AEP)	8704020263	3/27/87

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE	INDEX

Enclosure 3

USI Data Base Printout

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: D. C. COOK 2					
A-01	WATER HAMMER	/ /	NC		
A-02	ASYMMETRIC BLOWDOWN LOADS ON REACTOR PRIMARY COOLANT SYSTEMS	/ /	NC		LEAK BEFORE BREAK
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	02/28/89	C		
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		NO REQUIREMENTS
A-24	QUALIFICATION OF CLASS 1E SAFETY-RELATED EQUIPMENT	03/31/86	C		
A-26	REACTOR VESSEL PRESSURE TRANSIENT PROTECTION	12/31/77	C		LTOPS
A-31	RHR SHUTDOWN REQUIREMENTS	/ /	N/A		NEW PLANTS ONLY. SRP.
A-36	CONTROL OF HEAVY LOADS NEAR SPENT FUEL	04/10/86	C		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		SUBSUMMED BY A-46
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		SUBSUMED BY SEVERE ACC
A-46	SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS	/ /	I		REQ UNDER DEVEL
A-47	SAFETY IMPLICATIONS OF CONTROL SYSTEMS	03/31/90	E		NEW REQUIREMENTS
A-48	HYDROGEN CONTROL MEASURES AND EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT	12/31/90	I	ICE CONDENSER	UNDER NRC REVIEW
A-49	PRESSURIZED THERMAL SHOCK	01/22/86	C		

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: D. C. COOK 1					
A-01	WATER HAMMER	/ /	NC		
A-02	ASYMMETRIC BLOWDOWN LOADS ON REACTOR PRIMARY COOLANT SYSTEMS	/ /	NC		LEAK BEFORE BREAK
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	07/08/89	C		
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		NO REQUIREMENTS
A-24	QUALIFICATION OF CLASS 1E SAFETY-RELATED EQUIPMENT	07/31/85	C		
A-26	REACTOR VESSEL PRESSURE TRANSIENT PROTECTION	08/04/77	C		LTOPS
A-31	RHR SHUTDOWN REQUIREMENTS	/ /	N/A		NEW PLANTS ONLY. SRP.
A-36	CONTROL OF HEAVY LOADS NEAR SPENT FUEL	04/10/86	C		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		SUBSUMMED BY A-46
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		SUBSUMED BY SEVERE ACC
A-46	SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS	/ /	I		REQ UNDER DEVEL
A-47	SAFETY IMPLICATIONS OF CONTROL SYSTEMS	03/31/90	E		NEW REQUIREMENTS
A-48	HYDROGEN CONTROL MEASURES AND EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT	12/31/90	I	ICE CONDENSER	UNDER NRC REVIEW
A-49	PRESSURIZED THERMAL SHOCK	01/22/86	C		

SAIC-90/1038

POST-IMPLEMENTATION AUDIT REPORT FOR INDIANA MICHIGAN POWER COMPANY'S DONALD C. COOK NUCLEAR POWER PLANT UNITS 1 AND 2 SAFETY PARAMETER DISPLAY SYSTEM FEBRUARY 21 AND 22, 1990

TAC Nos. 61207 and 61208

March 20, 1990

Science Applications International Corporation

Prepared for:

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Washington, D.C. 20555

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POST-IMPLEMENTATION AUDIT REPORT FOR
INDIANA MICHIGAN POWER COMPANY'S
DONALD C. COOK NUCLEAR POWER PLANT UNITS 1 AND 2
SAFETY PARAMETER DISPLAY SYSTEM
FEBRUARY 21 AND 22, 1990

1.0 INTRODUCTION

The purpose of this report is to document the findings of a Nuclear Regulatory Commission (NRC) post-implementation audit of Indiana Michigan Power Company's Donald C. Cook Nuclear Plant, Units 1 and 2, Safety Parameter Display System (SPDS). The NRC team consisted of two staff members from NRC's Human Factors Assessment Branch and a Contractor from Science Applications International Corporation (SAIC). The purpose of the February 21 and 22, 1990 onsite audit was to assess the status of the SPDS with regard to eight NUREG-0737, Supplement 1, requirements (Reference 1).

The audit agenda is provided in Attachment 1. The list of meeting attendees is provided in Attachment 2, and licensee presentation materials are provided in Attachment 3.

2.0 BACKGROUND

The principle purpose and function of the SPDS is to aid control room personnel in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid degradation of the core. The SPDS can be particularly important during anticipated transients and the initial phase of an accident.

All holders of operating licenses and applicants must provide an SPDS in the control rooms of their plants. The NRC requirements for the SPDS are defined in NUREG-0737, Supplement 1. This document requires licensees and license applicants to prepare a written safety analysis report sufficient to assess the safety status of each identified function for a wide range of events, including symptoms of severe accidents. Licensees and applicants must also prepare an

implementation plan for the SPDS that contains schedules for design, development, installation, and full operation of the SPDS as well as a design verification and validation plan. The safety analysis report and the implementation plan were to be submitted to the NRC for staff review. The staff's review is published in a safety evaluation report.

The SPDS requirements, as defined by NUREG-0737, Supplement 1, are:

1. Provide a concise display of critical plant variables to control room operators (NUREG-0737, Supplement 1, Paragraph 4.1.a).

2. Be located convenient to control room operators (NUREG-0737, Supplement 1, Paragraph 4.1.b).

3. Provide continuous display of plant safety status information (NUREG-0737, Supplement 1, Paragraph 4.1.b).

4. Have a high degree of reliability (NUREG-0737, Supplement 1, Paragraph 4.1.b).

5. Be suitably isolated from electrical or electronic interference with safety systems (NUREG-0737, Supplement 1, Paragraph 4.1.b).

6. Be designed to incorporate accepted human factors engineering principles (NUREG-0737, Supplement 1, Paragraph 4.1.e).

7. Minimum information displayed shall be sufficient to determine plant safety status with respect to five safety functions (NUREG-0737, Supplement 1, Paragraph 4.1.f):

   i. Reactivity control
   ii. Reactor core cooling and heat removal from the primary system
   iii. Reactor coolant system integrity
   iv. Radioactivity control
   v. Containment conditions.

8. Implementation of procedures and operator training addressing actions with and without SPDS (NUREG-0737, Supplement 1, Paragraph 4.1.c).

Guidance for evaluating the application of the above requirements is provided by Appendix A to Section 18.2 of NUREG-0800 (Reference 2) and other documents cited therein, particularly NUREG-0700 (Reference 3).

On April 12, 1989, the NRC issued Generic Letter 89-06: "Task Action Plan Item I.D.2 - Safety Parameter Display System - 10 CFR Part 50.54(f)" (Reference 4). The letter requested all licensees of operating plants to perform an assessment and evaluation of the implementation status of their safety parameter display system. In addition, NUREG-1342, "A Status Report Regarding Industry Implementation of Safety Parameter Display Systems" dated April 1989 (Reference 5), was included with the generic letter to assist licensees in implementing SPDS requirements. The licensee responded to Generic Letter 89-06 by letter dated August 1, 1989 (Reference 6).

3.0 EVALUATION

The evaluation results are provided below in order of the eight NUREG-0737, Supplement 1, SPDS requirements.

3.1 Concise Display of Critical Plant Variables to Control Room Operators

The evaluation of concise display included a review of physical location of displayed information and technical information organization within the display screens.

The SPDS was displayed on five screens located near the shift supervisor's desk. The licensee defined the SPDS as the two top level iconic displays (see Attachment 3) along with a new Eberline digital radiation monitoring system and a containment isolation status system. Additional lower level displays were available on four of the SPDS screens, but were not part of SPDS. The combined technical content of the three SPDS display systems was sufficient to assess the status of the five NUREG-0737, Supplement 1, safety functions.

At the time of the audit, two parts of the SPDS were undergoing modifications and were defined by the licensee as non-operational. The "RAD" spoke on the iconics was non-operational because it was driven by Westinghouse instruments that were replaced by the new Eberline digital radiation monitoring system. Second, the new containment isolation status display system was not operational because it was undergoing final modification and testing. These systems were scheduled for operation following the 1990 refuelling outages.

It was the review team's judgment that the licensee will meet the NUREG-0737, Supplement 1, requirement for a concise display when the Eberline radiation monitoring and the digital containment isolation status systems are completed and declared operational.

3.2 Located Convenient to Control Room Operators

The evaluation of SPDS workspace location included an assessment of how the SPDS displays and controls supported the operator's needs during emergency operations. This included a determination of who was, defined by the licensee, a user of SPDS.

The SPDS components consisted of four screens and keyboards that were used to display the iconics, and the containment isolation valve status system, and one screen with keyboard for the Eberline radiation monitoring system. The five display screens were set up near the shift supervisor's desks in front of the control room. All screens and keyboards were convenient for use by the shift supervisor or shift technical advisor during emergency operations.

It was the review team's judgment that the licensee met the NUREG-0737, Supplement 1, requirement for a display convenient to control room operators.

3.3 Continuous Display of Plant Safety Status Information

The team evaluated the SPDS to determine if it continuously displayed information about the five critical plant variables identified in NUREG-0737, Supplement 1.

The licensee limited its definition of SPDS to the two (wide and narrow range) iconic displays, the containment isolation status display, and the Eberline radiation monitoring system display. This definition excluded the lower level displays available on the SPDS screens.

The wide range SPDS iconic screen was designed to be displayed automatically on a reactor trip signal. This iconic provided an overview status of all five NUREG-0737, Supplement 1, safety functions. However, in emergency operations, it would be necessary for the operators to call up the lower level detail displays for information not available on the iconic displays. This would result in loss of the iconics. The licensee did not have an administrative or technical system designed to maintain a dedicated iconic display of plant status.

It was the review team's judgment that the licensee did not meet the NUREG-0737, Supplement 1, requirement for a continuous display because lower level detail displays could replace the overview iconic displays.

3.4 Should Have a High Degree of Reliability

In order to be judged reliable, SPDS should have a high degree of hardware and software availability.

In the Generic Letter 89-06 checklist, the licensee stated that over the preceding 12 months the SPDSs in both units had achieved greater than 99% availability. During the audit the iconic displays, Eberline radiation monitoring systems and containment isolation status systems in both units were operating reliably.

It was the review team's judgment that the licensee met NUREG-0737, Supplement 1, requirement for a high degree of reliability.

3.5 Suitably Isolated from Electrical and Electronic Interference with Safety Systems

The licensee's electrical and electronic isolation was evaluated previously by the NRC and found to be acceptable in an SER dated August 12, 1985.

3.6 Designed Incorporating Accepted Human Factors Engineering Principles

The review team evaluated the human factors aspects of SPDS in the control room, including the SPDS technical content, display formats, and workstation designs.

The technical content of the iconic displays was found to be generally acceptable. However, the review team did identify two plant-specific concerns associated with the iconic displays.

First, the Unit 2 iconic spoke for narrow range low steam generator level was set at 17%, which was correct for Unit 1 but not for Unit 2. Each unit had its own operating characteristics. Unit 2 low steam generator level setpoint was 21% and Unit 1 was 17%. The Unit 2 SPDS setpoint should have been modeled on Unit 2 rather than Unit 1.

Second, the Unit 1 Technical Specifications for high T-Avg was changed in June 1989, but the SPDS was not modified to reflect the change. The high Unit 1 T-Avg should have been changed from 571.8°F to 570.9°F.

These discrepancies were not apparent to the operators because the numerical values for the high and low level alarm setpoints on each parameter spoke of the narrow and wide range iconic displays were not displayed. Some alarm setpoints were reactor trip values and others were either anticipatory values or system capacity values.

The formats of the iconic displays were reviewed and found acceptable. The review team evaluated the formats for the containment isolation valve status system and Eberline Radiation Monitoring System and found them acceptable.

The workstation design included five display screens and keyboards arranged on the shift supervisor's desk. This was a relatively large number of screens and keyboards. However the licensee did make an effort to consolidate SPDS information in a central location. The review team found the workstation design acceptable.

It was the review team's judgment that the licensee did not meet the NUREG-0737, Supplement 1, requirement for a display incorporating accepted human factors principles because of the incorrect parameter setpoints discussed above.

3.7 Minimum Information Displayed Should be Sufficient to Determine Plant Safety Status With Respect to Five Safety Functions

The SPDS parameters used to depict the five NUREG-0737, Supplement 1, critical safety functions are listed below.

NUREG-0737, Supplement 1 Safety Functions	Parameters
1. Reactivity Control	Neutron Flux
	x Source Range
	x Intermediate Range
	x Power Range
	x Other: Turbine Impulse Pressure
2. Reactor Core Cooling and Heat Removal from the Primary System	x RCS Level
	x Subcooling Margin
	___ Hot Leg Temperature
	___ Cold Leg Temperature
	x Core Exit Thermocouples
	x Steam Generator Level
	___ Steam Generator Pressure
	___ RHR Flow
	x Other: PRZR Pressure
	RCS Tavg
	RCS Pressure

NUREG-0737, Supplement 1 Safety Functions	Parameters
3. RCS Integrity	x RCS Pressure ___ Cold Leg Temperature x Containment Sump Level x Steam Generator Level x Other: PRZR Level Wide & Narrow Range Net Charging Flow a. RCP 1 seal return flow b. RCP 2 seal return flow c. RCP 3 seal return flow d. RCP 4 seal return flow e. Letdown flow Radiation Monitoring Containment Temperature maximum Containment Pressure RV Level
4. Radioactivity Control	x Stack Monitor(s) x Steamline Radiation(s) x Containment Radiation x Other: Effluent Monitors Plant Unit Vent, Airborne, Steam Jet Air Ejector Noble Gas, Gland Seal, Condensor Exhaust Noble Gas, Main Steam Relief for Atmosphere, (SG PORV Monitor) Gross Gamma, Steamline Radiation
5. Containment Conditions	x Containment Pressure x Containment Isolation ___ Containment Hydrogan Concentration x Other: Containment Sump Level inputs Containment temperature inputs Radiation Monitoring inputs

Residual heat removal (RHR) system flow and containment hydrogen concentration were not included in licensee's list of SPDS parameters. The review team determined that RHR flow in gallons per minute (GPM) was available on lower level display 3RC1-RCS, (see Attachment 3). Containment hydrogen concentration was also available on a lower level screen and alarmed at 4% on an

annunciation tile. Both RHR flow and containment hydrogen could be displayed when needed.

It was the review team's judgment that the licensee met the NUREG-0737, Supplement 1, requirement for minimum information sufficient to determine plant safety status with respect to five safety functions.

3.8 Procedures and Operator Training Addressing Operator Actions With and Without SPDS

The SPDS training consisted of less than 2 hours classroom, with no periodic update training. Interviews with operators and shift technical advisors revealed that they did not have a working knowledge of detailed SPDS information such as high and low setpoints on the iconic spokes.

It was the review team's judgment that the licensee did not meet the NUREG-0737, Supplement 1, requirement for procedures and training addressing operator actions with and without SPDS.

4.0 CONCLUSIONS

The purpose of the NRC's February 21 and 22, 1990 post-implementation audit of Indiana Michigan Power Company's Donald C. Cook Nuclear Plant, Units 1 and 2, Safety Parameter Display System was to assess the status of the system with regard to the eight NUREG-0737, Supplement 1, requirements. As a result of the audit, it was the review team's judgment that the licensee met five of the eight requirements.

The licensee did not meet the requirement for continuous display of plant safety status information because the overview status provided by the iconic displays could be replaced with lower level displays. The requirement for incorporation of accepted human factors principles was not satisfied because two parameter alarm setpoints were incorrect. Also, the requirement that operators should be trained with SPDS was not met because operators did not have working

knowledge of the high and low level alarm setpoints for each parameter on the narrow and wide range iconic displays.

The licensee committed to evaluate the NRC's concerns and respond.

REFERENCES

1. NUREG-0737, Supplement 1, Requirements for Emergency Response Capability, Generic Letter 82-33, NRC, December 7, 1982.

2. NUREG-0800, Standard Review Plan of Safety Analysis Report for Nuclear Power Plants, Section 18.2, Rev. 0, Safety Parameter Display System (SPDS), Appendix A to SRP Section 18.2, NRC, November 1984.

3. NUREG-0700, Guidelines for Control Room Design Reviews, NRC, September 1981.

4. Generic Letter 89-06: Task Action Plan Item I.D.2 - Safety Parameter Display System - 10 CFR 50.54(f), NRC, April 12, 1989.

5. NUREG-1342, A Status Report Regarding Industry Implementation of Safety Parameter Systems, NRC, April 1989.

6. Generic Letter 89-06 Task Action Plan Item I.D.2 Safety Parameter Display System (SPDS) Response, Indian Michigan Power Company, August 1, 1989.

ATTACHMENT 1
AGENDA

TENTATIVE AGENDA FOR
INDIANA MICHIGAN POWER COMPANY'S
D.C. COOK - UNITS 1 AND 2
SAFETY PARAMETER DISPLAY SYSTEM AUDIT
(February 21 through 23, 1990)

Wednesday, February 21, 1990

1:00 p.m. NRC Safety Parameter Display System (SPDS) Entrance Briefing
- SPDS Generic Letter 89-06 status
- Previous NRC findings regarding the D.C. Cook SPDS

1:15 p.m. Licensee presentation on SPDS

2:00 p.m. Control room and/or simulator orientation and begin SPDS discussion

5:00 p.m. End Day 1

Thursday, February 24, 1990

8:30 a.m. SPDS discussion (continued)

- Completeness of parameters selected to display information about:

  1. Reactivity control
  2. Reactor core cooling and heat removal from the primary system
  3. Reactor coolant system integrity
  4. Radioactivity control
  5. Containment conditions

- Generic Letter 89-06 (NUREG-1342)/D.C. Cook differences

10:00 a.m. NUREG-0737, Supplement 1 requirements for:

1. Concise display of critical plant variables
2. Located convenient to control room operators
3. Should have a high degree of reliability
4. Continuous display of critical plant variables
5. Designed incorporating accepted human engineering principles.
6. Procedures and operator training addressing actions with and without SPDS.

12:00 Lunch

1:00 p.m. NUREG-0737, Supplement 1 (continued)

2:00 p.m. NRC caucus

3:00 p.m. Open issues technical discussion with licensee/pre-exit briefing

Friday, February 25, 1990

8:30 a.m. Exit briefing

ATTACHMENT 2
LIST OF ATTENDEES

Attendees at the D.C. Cook - SPDS Audit Exit Meeting

R. Correia	NRC Technical Reviewer for SPDS Sr. Op. Engineer
C. Goodman	NRC Sr. H.F. Specialist
J. DeBor	Contractor (SAIC)
B. Stoner	Computer Science
J. Allard	Computer Science
B. Jurgensen	NRC/Sr. Res. Insp.
T. Kobetz	NRC/Reactor Engineer
J. Paris	Computer Science, ACC
G. Hageniers	Computer Science
R. Stephens	Operations
R. B. Gridley	NED/Engineer
W.T. Rae	NOD/Engineer
K.J. Toth	NOD/Licensing

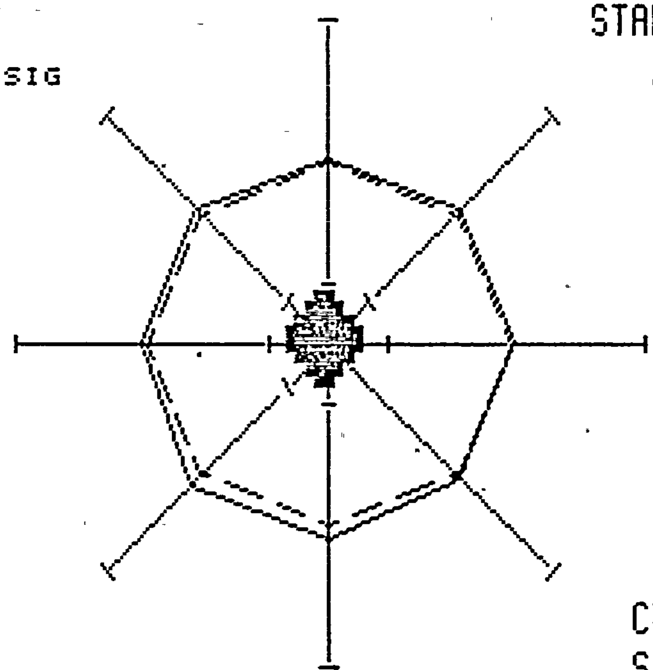
ATTACHMENT 3
LICENSEE'S PRESENTATION MATERIAL

1TL2 WID RNG | D C COOK UNIT 1 | 22 FEB 90 14:39:30

CORE EXIT
604 / 620 °F
40 SUBCOOL

WID RCS PRESS
2093 / 2085 PSIG
292 MOP

STARTUP
.0 DPM



PRZR LEV
40 / 40M %

CNTMT PRESS
.1 PSIG

RV LEV / EXP
108 / 100 %

RAD
CNTMT NORM
SEC NORM
OTHER NORM

WID SG LEV LP 1
57 / 55 %

ENTER POINT TAG NUMBER

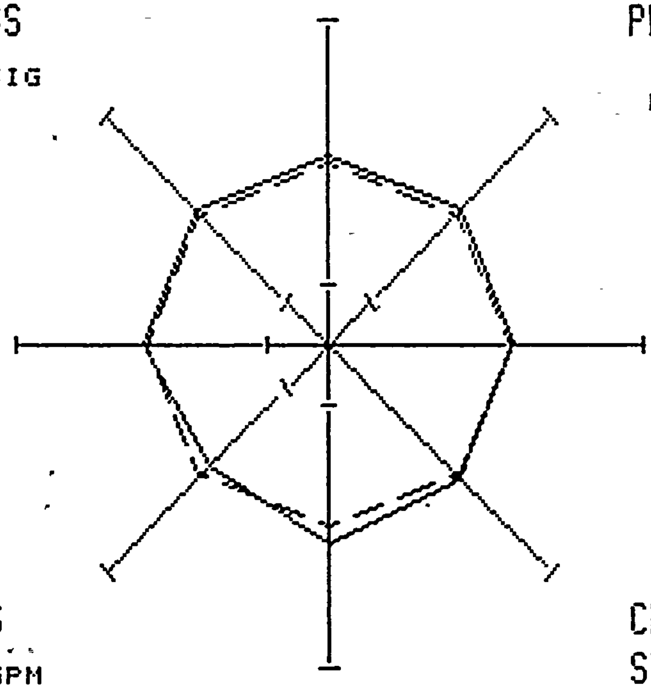
INSTRUCTIONS
1. TYPE IN POINT TAG NO.
2. DEPRESS ENTER

1TL1 NAR RNG • PSSD D C COOK UNIT 2 22 FEB 90 14:40:03

TAVG LP 1
572.6 / 573.5 °F

PRZR PRESS
2234 / 2235 PSIG

PWR MISMATCH
100 / 100 %
NUC TURB



PRZR LEV
54 / 53 %

CNTMT
TEMP NORMP
SUMP LEV NORM

RAD
CNTMT NORM
SEC NORM
OTHER NORM

NET CHG
-6 GPM

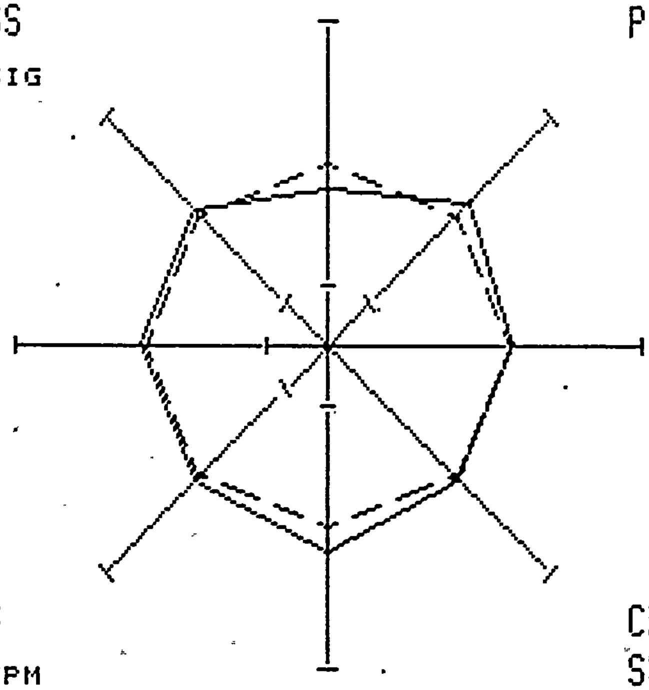
NAR SG LEV LP 4
46 / 44 %

1TL1 NAR RNG PSSD D C COOK UNIT 1 22 FEB 90 14:39:01

Tavg LP 3
548.3 / 549.9 °F

PRZR PRESS
2090 / 2085 PSIG

PWR MISMATCH
101 / 100 %
NUC TURB



PRZR LEV
40 / 40M %

CNTMT
TEMP NORM
SUMP LEV NORM

NET CHG
2 GPM

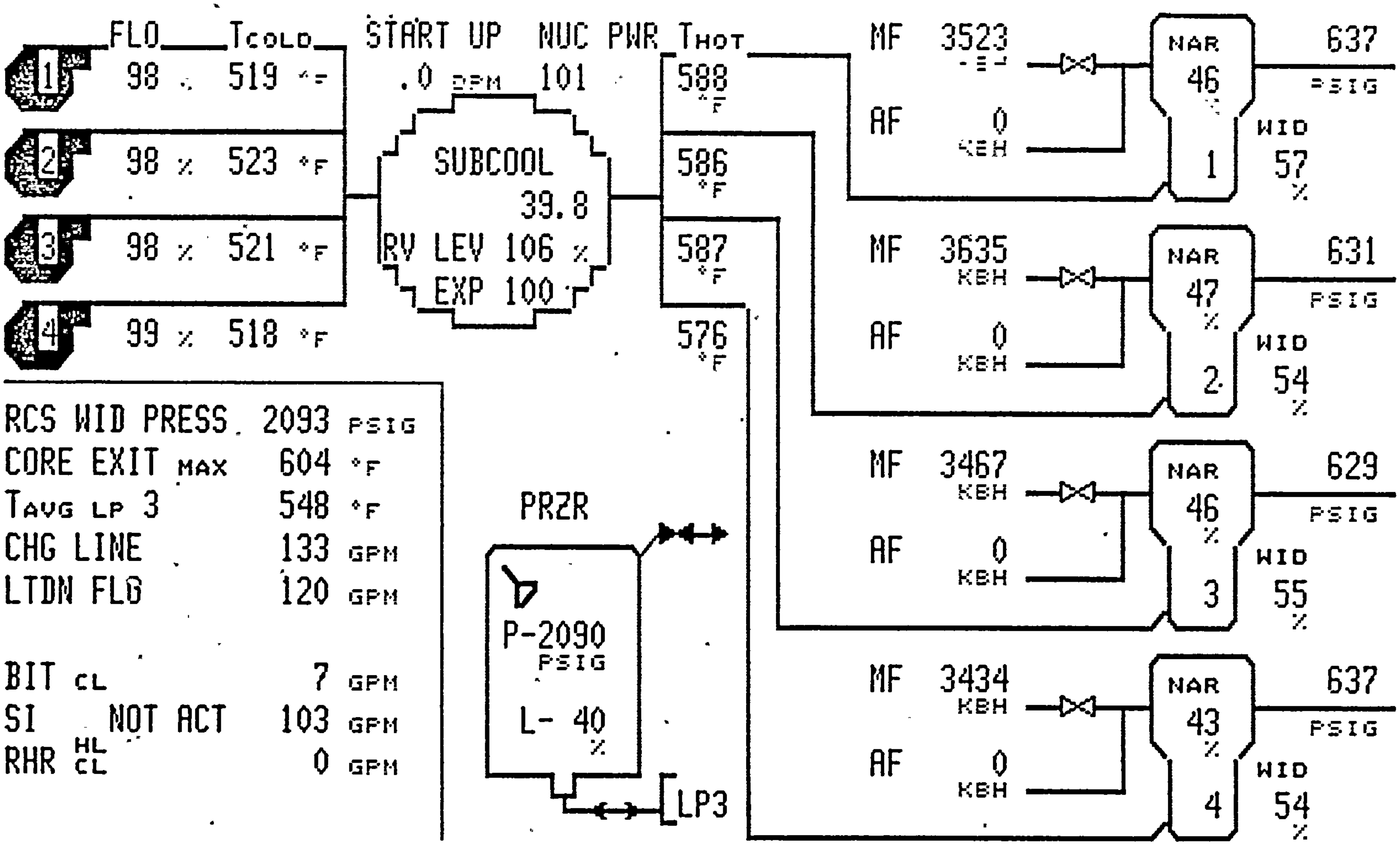
NAR SG LEV LP 1
47 / 44 %

RAD
CNTMT NORM
SEC NORM
OTHER NORM

ENTER POINT TAG NUMBER

INSTRUCTIONS
1. TYPE IN POINT TAG NO.
2. DEPRESS ENTER

3RC1 RCS NDL D C COOK UNIT 1 22 FEB 90 14:37:43



ENTER DIAGRAM NO.

MANUAL REQUEST

INSTRUCTIONS
1. TYPE IN DIAGRAM NO.
2. DEPRESS EXECUTE