UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 5, 1998

MEMORANDUM TO: Thomas H. Essig, Acting Chief
Generic Issues and Environmental Projects Branch
Division of Reactor Prog.am Management, NRR

FROM: Michael J. Davis, Project Manager
Generic Issues and Environmental Projects Branch
Division of Reactor Program Management, NRR

Michael J. Davis

SUBJECT: SUMMARY OF PUBLIC MEETING HELD ON APRIL 13, 1998, WITH THE BOILING WATER REACTOR OWNERS GROUP (BWROG) APPENDIX R COMMITTEE

On April 13, 1998, the staff held a public meeting with representatives of the Boiling Water Reactor Owners Group (BWROG) Appendix R Committee at NRC Headquarters in Rockville, Maryland. The purpose of the meeting was to discuss proposed generic guidelines on safe shutdown circuit analysis. A list of attendees and their affiliations is provided as Attachment 1. A copy of the handout used in the BWROG presentation is provided as Attachment 2. A copy of the handout used in the NRC presentation is provided as Attachment 3.

A BWROG representative began the meeting with a brief update on the Appendix R Committee's activities. The committee was formed during the last quarter of 1997 to look into generic Appendix R fire protection issues. The charter was recently expanded to include safe shutdown circuit analysis issues and tasked the committee to improve communications with the NRC in this area. Each of the BWR owners have participated in Appendix R Committee meetings and Nuclear Energy Institute representatives attend all meetings also. Contact with other owners groups is primarily through executive committee members, who keep the other owners groups informed of BWROG initiatives.

The BWROG Appendix R Committee is developing a circuit analysis document called "Generic Guidance for Appendix R Circuit Analysis Assessments." A copy of the draft table of contents was attached to the agenda (see Attachment 2).

The staff distributed copies of the NRC Circuit Analysis Resolution Plan (see Attachment 3), which included a compilation of recent deficiencies noted in the area of circuit analysis and a schedule for issuance of an Information Notice and Generic Letter (GL) on that topic. The inspection process uncovered a potential generic issue about a year ago in the area of safe shutdown circuit analysis. Enforcement guidance was issued by the staff in March 1998. The Generic Letter will identify and consolidate the regulatory requirements in the area of circuit analysis and attempt to resolve industry questions. The staff is not trying to impose any new requirements, but interpret what already exists. The GL will provide clarification and require licensees to look at their own programs to verify that they were performed correctly. If licensees find problems, they will be resolved using the normal notification and corrective action programs. There is also a tentative plan for NRC followup inspections.

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A BWROG representative commented that more interaction with the industry was needed before the draft GL is issued. The staff is considering the possibility of a workshop to give all the industry groups an opportunity to interact with the staff in an efficient manner on this subject. The circuit analysis document being prepared by the Appendix R Committee will have an owners group position for each of the various issues. The staff reacted positively to the suggestion that the BWROG document might be submitted for staff review. The staff felt it would be beneficial for the BWROG to submit information on issues as they are identified, which would state the requirement and how the BWROG would intend to meet each requirement.

The next Appendix R Committee meeting is scheduled for May 5 and 6, 1998, in Orlando, Florida. It was suggested that the BWROG could extend an invitation for the NRC staff to attend this meeting. The staff will brief NRC management on this proposal and get back to the BWROG with a response.

Project No. 691

Attachments: As stated

cc w/ atts: See next page

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LIST OF ATTENDEES AT MEETING WITH BWROG HELD IN
ROCKVILLE, MARYLAND ON APRIL 13, 1998

NAME	AFFILIATION
M. Davis	NRC
S. West	NRC
M. Straka	NUS Information Service
A. Wyche	Bechtel
D. Robare	GE
L. Whitney	NRC
P. Madden	NRC
B. Matthews	Southern Nuclear
C. Bajwa	NRC

Attachment 1

NRC/BWROG APPENDIX R MEETING

APRIL 13, 1998 2:30PM

AGENDA

Introductions

BWROG Committee Update

- Charter
- Membership
- NEI Relationship
- Joint Owner's Group Relationship
- Circuit Analysis Report

NRC Appendix R Plan

Future NRC/BWROG Interactions

Attachment 2

DRAFT BOILING WATER REACTOR OWNERS' GROUP

GENERIC GUIDANCE FOR APPENDIX R CIRCUIT ANALYSIS ASSESSMENTS

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GENERIC GUIDANCE FOR APPENDIX R CIRCUIT ANALYSIS ASSESSMENTS

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April 7, 1998, Rev. 6

CIRCUIT ANALYSIS RESOLUTION PLAN

1 DESCRIPTION

This resolution plan is used to track and manage the staff's actions to resolve apparent widespread noncompliance with regulatory requirements that specify that facilities be designed such that fire-induced circuit failures (e.g., hot shorts, open circuits, and shorts to ground) will not adversely impact the ability to achieve and maintain safe shutdown. Principal actions include an information notice followed by a generic letter. A table of work activities, milestones, schedules, and status is included as Attachment 1. Some of the activities described in the table will be conducted in parallel.

TAC Number:	M96913 (CARP and generic letter) MA1361 (information notice)
DSSA Operating Plan:	Subactivity 11J1, Strategy 1.1.3
Responsible Organization:	Fire Protection Engineering Section Plant Systems Branch Division of Systems Safety and Analysis Office of Nuclear Reactor Regulation

2 BACKGROUND

The underlying purpose of the regulatory requirements of interest is to ensure that systems and associated circuits used to achieve and maintain safe shutdown are free from fire damage (45 FR 76602). The Office of Nuclear Reactor Regulation (NRR), the regional offices, and licensees have found plant-specific problems related to potential fire-induced electrical circuit failures which could prevent operation or cause maloperation of equipment needed to achieve and maintain post-fire safe shutdown. An example is the type of problem documented in Information Notice (IN) 92-18, "Potential For Loss of Remote Shutdown Capability During a Control Room Fire," February 28, 1992. This IN informed licensees that the circuit logic associated with certain motor-operated valves, when subjected to a single fire-induced hot short, could result in a spurious permissive signal. The spurious signal could cause the valve to operate, bypassing the protective features, resulting in mechanical valve damage. Such fire-induced damage could impair the capability to shut down the plant and maintain it in a safe shutdown condition.

Inspections conducted in each region have found that there may exist widespread noncompliance with regulatory requirements and problems with licensee implementation of regulatory guidance and staff positions that specify that facilities be designed such that fire-induced circuit failures (e.g., hot shorts, open circuits, and shorts to ground) will not adversely impact the ability to achieve and maintain a safe shutdown condition. The noncompliance

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Leon Whitney, NRR/SPLB 301-415-3081

appears to stem, in part, from misunderstanding of the requirements and inappropriate reliance on a 1992 recommendation from the Nuclear Management and Resources Council (NUMARC), which was not endorsed by the NRC, and may have caused confusion during licensee evaluations. Poor understanding of the design requirements has also been cited as a contributing factor. Licensee event reports (LERs), and information submitted by the Nuclear Energy Institute (NEI, formerly NUMARC) have also revealed that some licensees have failed to properly identify, analyze, and protect electrical circuits required to achieve and maintain post-fire safe shutdown. Associated non-safety circuits appear to be subject to the same problems.

In a letter of January 14, 1997, NEI raised concerns regarding IN 92-18 and suggested that the circumstances discussed in IN 92-18 were not consistent with the intent of the fire protection regulation. During a public meeting on February 7, 1997, the staff discussed with NEI the questions and issues it raised in its letter. Later, in its response to NEI dated March 11, 1997, the staff reiterated its position that the potential for fire-induced circuit failures to impair the capability to achieve and maintain safe shutdown was within the scope of the existing fire protection regulation. Although the letter focused on NEI's questions regarding IN 92-18, its enclosure explained staff positions germane to the larger issue of fire-induced circuit failures.

In a follow-up letter of May 30, 1997, and later in a meeting of June 4, 1997, NEI informed the staff that the positions held by the majority of the licensees differed from the staff positions. As a result of this new information, the staff expanded the scope of the issue from IN 92-18 to the entirety of post-fire safe shutdown circuit analyses.

A summary of recent inspection findings and LERs related to post-fire safe shutdown circuit analyses is included as Attachment 2.

3 SCOPE OF WORK

To resolve the apparent widespread noncompliance with the regulatory requirements, the staff will alert licensees to potential problems associated with fire-induced circuit failures and spurious actuations, will clarify the regulatory requirements by consolidating existing staff positions regarding fire-induced circuit failures and post-fire safe shutdown capability, and will require the licensees to either confirm that they meet the requirements or, if they do not, to implement corrective actions. The staff, in its letter to NEI of March 11, 1997, explained the regulatory requirements and summarized how existing staff positions and regulatory guidance implement them. In the short term, as a supplement to that letter, the staff will issue an IN to alert licensees to recent plant-specific problems with post-fire safe shutdown circuit analyses. Following the IN, in the near term, the staff will issue a generic letter that will consolidate existing staff positions and clarify the regulatory requirements. The generic letter will establish a time frame for licensees that do not meet the regulatory requirements to implement permanent corrective actions.

To support these actions, the staff will review operating experience and NRC inspection results; assess the data and information obtained from the document reviews; identify commonality or correlation of evidence, if any, that suggest widespread plant-specific problems, or generic concerns of safety significance; determine if NRC requirements, review guidance, and inspection procedures are adequate; and determine if existing industry programs are adequate. The staff is coordinating this work with fire protection functional inspection and performance-based, risk-informed fire protection rulemaking activities. The staff will determine the appropriate follow up activities (e.g., inspections) after it issues the generic letter.

The staff will also provide input to the Office of Enforcement for the preparation of an Enforcement Guidance Memorandum (EGM). The EGM will provide guidance regarding violations of regulatory requirements that specify that facilities be designed such that fire-induced circuit failures will not preclude the ability to achieve and maintain post-fire safe shutdown.

4 JUSTIFICATION FOR ADDRESSING ISSUE

4.1 Regulatory Assessment

Each reactor has an NRC-approved fire protection program that, if properly designed, implemented, and maintained, satisfies Section 50.48, "Fire protection," and General Design Criterion 3, "Fire protection," of Title 10 of the *Code of Federal Regulations*, Part 50. These programs use the "defense in depth" concept of echelons of fire protection to achieve a high degree of fire safety. The objective of the concept is to (1) prevent fires from starting; (2) promptly detect, control, and extinguish those fires that do occur; and (3) protect structures, systems, and components important to safety so that a fire that is not promptly extinguished will not prevent the safe shutdown of the plant. A deficient post-fire safe shutdown circuit analysis would be a weakness in the third echelon of fire protection defense in depth. Such deficiencies could adversely impact the ability to achieve and maintain safe shutdown in the event of a fire. However, the multiple layers of fire protection produced by the defense-in-depth concept offer reasonable assurance that weaknesses or deficiencies in one echelon will not present an undue risk to public health and safety. Therefore, there is reasonable assurance that a fire will not present an undue risk to public health and safety while the staff carries out this resolution plan and the licensees implement corrective actions. (Examples of corrective actions include reanalysis, cable rerouting, and fire barrier installations. It should also be noted that if plant-specific deficiencies are found, interim compensatory measures may be warranted until corrective actions are completed.) Nevertheless, the regulatory requirements regarding post-fire safe shutdown capability have safety benefit, the deficiencies are noncompliances that may reduce fire protection safety margins, and safety improvements can be made within the existing regulation by the licensees. Therefore, completion of this resolution plan is the appropriate staff action for this issue and the identification and correction of circuit problems is the appropriate licensee response to this issue.

4.2 Risk Considerations

According to the "Fire Risk Scoping Study" performed by Sandia National Laboratories, plant modifications made as a result of Appendix R to 10 CFR Part 50 reduced core damage frequencies (CDFs) at some plants by a factor of ten. However, risk assessments still show that fire events are risk significant and that fire-induced CDFs can be comparable to internal events (above 10E-5/reactor year) and that fires can contribute more than 50 percent to the plant's overall CDF. (Estimates are generally in the range of 20 to 40 percent.)

Deficiencies associated with circuit analyses may cut into the gains achieved by Appendix R compliance. Work that is currently being performed by Brookhaven National Laboratory (BNL) provides some insights into the risk significance of the circuit analysis issue. For example, BNL draft Technical Letter Report, "Self-Induced Station Blackout: Procedural Review and Risk Evaluation," dated August 27, 1997, states the following:

- Failure to protect certain electrical cables from circuit failure may affect the capability to achieve and maintain post-fire safe shutdown due to spurious actuations of valves and equipment. Given the risk that spurious actuations may occur before emergency diesel generator (EDG), selected bus, and supporting equipment isolation transfer switches are actuated, the maximum core damage frequency (CDF) for alternative shutdown panel design reactor plants for a fire involving the unprotected circuits is point estimated at 6.5E-5/reactor year.

- Self-initiated station blackout (SISBO) plants (those without switches for transferring control to alternative shutdown locations) may also experience spurious actuations before SISBO implementation. Additionally, the SBO may create conditions which may lead to core uncovery in one and one-half hours (such as having a power operated relief valve [PORV] stuck open). For these plants the maximum CDF point estimate is 1.5E-4/reactor year, which is greater than the NRC's subsidiary safety goal for risk from all internal and external events of 1E-4/reactor year.

As part of its work on this issue, the staff will consider additional risk information as appropriate.

4.3 Conclusions

On the basis of its regulatory assessment and the risk insights, the staff concluded that the circuit analysis issue is a potentially safety and risk significant issue that requires near term staff evaluation (NRR review priority 2).

5 REFERENCES

5.1 Regulatory Requirements:

Generic Letter (GL) 81-12, "Fire Protection Rule," February 20, 1981

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.48, "Fire protection."

10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979."

- Section III.G. "Fire Protection of Safe Shutdown Capability"
- Section III.L. "Alternative and Dedicated Shutdown Capability"

5.2 Staff Review Guidance

NUREG-0800, "Standard Review Plan," Section 9.5-1, "Fire Protection Program."

Branch Technical Position CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants"

5.3 Staff Positions

Generic Letter (GL) 86-10, "Implementation of Fire Protection Requirements," April 24, 1986

- Interpretation 3 defines the term "free of fire damage" in terms of ability to perform intended function.

- Response 3.8.4 "Control Room Fire Considerations" (addresses physical and electrical independence of alternative or dedicated safe shutdown capability)
- Response 5.3.1. "Circuit Failure Modes"
- Response 5.3.2. "Hot Short Duration"
- Response 5.3.10 "Design Basis Plant Transients" (addresses single versus multiple spurious actuations and/or signals, loss of automatic functions, and spurious actuation of redundant high-low pressure interface valves)

USNRC, memorandum of January 16, 1991, to T. Martin, from B. Boger, Subject: "Request for assistance: determine whether two hot shorts in a multiconductor cable associated with a non-hi/low pressure interface should be analyzed for fire induced spurious actuation (Generic Letter 86-10, Section 5.3.1., non-hi/low pressure interfaces in ungrounded AC and DC circuits)."

5.4 Additional References

USNRC, IN 92-18, "Potential For Loss of Remote Shutdown Capability During A Control Room Fire," February 28, 1992.

NUMARC, letter of August 13, 1992, to NUMARC Administrative Points of Contact, Subject: NRC Information Notice (IN) 92-18, Potential For Loss of Remote Shutdown Capability During A Control Room Fire."

NEI, letter of January 14, 1997, to F.J. Miraglia, Jr., USNRC.

USNRC, memorandum of March 7, 1997, to D.B. Matthews, from M. Malloy, Subject: "Summary of February 7, 1997, Meeting With The Nuclear Energy Institute (NEI) on Information Notice 92-18."

USNRC, letter of March 11, 1997, to R.E. Beedle, NEI.

NEI, letter of May 30, 1997, to L.B. Marsh, USNRC.

USNRC, note of August 26, 1997, to S.J. Collins, from G.M. Holahan, Subject: "Action Plan, Post-Fire Safe Shutdown Circuit Analyses."

USNRC, memorandum of December 2, 1997, to T.H. E sig, from M. Malloy, Subject: "Summary of June 4, 1997, Meeting With The Nuclear Energy Institute (NEI) on Information Notice 92-18."

CIRCUIT ANALYSIS RESOLUTION PLAN SCHEDULE

Task No.		Task Description	Schedule	Status
1.	**Regional Guidance**			
	a.	Meet with Office of Enforcement (OE)	As needed	Complete
	b.	Provide input for Enforcement Guidance Memorandum (EGM) to OE	12/24/97 01/13/98	Complete
	c.	Issue EGM (OE)	01/30/98	03/02/98 C
2.	**Document Collection and Review**			
	a.	Obtain inspection reports issued during 1997 that address circuit analysis issues.	12/01/97	Complete
	b.	Obtain licensee event reports issued during 1997 that address circuit analysis issues.	12/01/97	Complete
	c.	Obtain documentation of staff positions regarding circuit analyses that have been issued since Appendix R was issued.	12/30/97	Complete
	d.	Assess documentation obtained under Tasks 2.a., 2.b., and 2.c. for use in developing information notice and generic letter.	12/30/97	Complete
3.	**Information Notice**			
	a.	Develop draft IN (include authorization form draft IN, excerpts from supporting references). IN will alert licensees to potential problems associated with post-fire safe shutdown circuit analyses. It will include discussions of recent inspection results, enforcement actions, and staff resolution plan.	02/20/98	02/20/98 C
	b.	Deliver draft IN to PECB for processing.	02/23/98	03/20/98 C
	c.	Resolve PECB comments.	03/31/98	03/31/98 C
	d.	PECB issue IN	04/30/98	

Task No.	Task Description		Schedule	Status
4.	**Generic Letter**			
	a.	Distribute first draft GL to HQ and regional staff for comment. The draft generic letter will clarify regulatory requirements regarding post-fire safe shutdown analyses. It will establish a time frame for licensees that are in noncompliance to implement permanent corrective actions.	05/29/98	
	b.	Receive HQ and regional staff comments on first draft GL.	06/19/98	
	c.	Resolve HQ and regional staff comments, develop proposed GL and CRGR package.	07/03/98	
	d.	Obtain and resolve technical editor comments on proposed GL.	07/17/98	
	e.	- Route proposed GL and CRGR package within DSSA for review and concurrence. - Route proposed GL and CRGR package to OE for review and concurrence. - Route proposed GL and CRGR package to OGC for review and no legal objection.	07/31/98	
	f.	Present proposed GL to ACRS.	08/14/98	
	g.	Resolve ACRS comments. Present proposed GL to CRGR.	08/28/98	
	h.	Resolve CRGR comments. Develop public comment GL. Develop Federal Register Notice and issue GL for public comment.	09/25/98	
	I.	60 day GL public comment period ends.	11/24/98	
	j.	Resolve public comments on GL. Develop redline/strikeout version of GL. Present redline/strikeout version of GL to CRGR.	12/22/98	

Task No.	Task Description	Schedule	Status
	k. Resolve CRGR comments on redline/strikeout version of GL. Develop Final GL.	01/18/99	
	l. Develop Commission Information Paper on Final GL. Submit Paper and Final GL to Commission.	02/19/99	
	m. Issue Final Generic Letter	03/5/99	
5.	**Followup Inspection (Tentative Plan)**		
	a. Verify inspection followup resource availability. Develop inspection guidance.	May, 1999	
	b. Train regional inspectors.	June, 1999	
	c. Commence regional sample inspections.	When GL licensee action period has expired.	
6.	**Industry Interaction and Coordination**		
	a. Public meeting with NEI to discuss staff concerns and issues.	06/04/97	Complete
	b. Present overview of proposed resolution plan to licensees and NEI.	10/15/97 (NEI Fire Protection Information Forum)	Complete
	c. Present overview of CARP to licensees and NEI.	03/23/98 (NEI Fire Protection Information Forum)	Complete
	d. Meet with BWROG.	04/13/98	
	e. Hold industry workshop(s) on Final GL (optional).	mid-June, 1999	

RECENT CIRCUIT ANALYSIS OPERATING EXPERIENCE

Region I

On July 26, 1996, Consolidated Edison Company reported that for certain fire scenarios at Indian Point Unit 2 the alternative safe shutdown instrumentation in the containment and in the primary auxiliary building fan house could be lost. The licensee determined that this could be caused by a fire which would affect the normal air supply as well electric power to the solenoid valve associated with the backup nitrogen supply (refer to 50.72 Immediate Notification 30790).

On March 29, 1996, Consolidated Edison Company reported that its Indian Point Unit 2 Appendix R re-analysis program had determined that a cable tray fire (common enclosure concern) in a certain fire area might result in a DC control power hot short. This could have resulted in the anticipated motion of the pressurizer PORVs and/or the block valves during a fire (refer to 50.72 Immediate Notification 30203).

On April 7, 1997 Niagra Mohawk Power company reported that at Nine Mile Point Unit 2, that portions of the control circuits for the Division I and II emergency diesel generator cooling water outlet valves were not isolated from the main control room. Therefore, a fire in the main control room had the potential to cause a disabling control circuit short and the loss of remote shutdown panel controls for these valves (refer to 50.73 Licensee Event Report 50-410/97-002).

On January 9, 1998, Boston Edison Company reported that at Pilgrim motor operated shutdown cooling suction isolation valves were vulnerable to mechanical damage from fire-induced hot shorts. The licensee reported that damage to the valves could cause unavailability of the shutdown cooling mode of the residual heat removal system during post-fire safe shutdown (refer to 50.72 Immediate Notification 33512 and 50.73 Licensee Event Report 50-293/97-029).

On February 13, 1996, the NRC cited Vermont Yankee Nuclear Power Corporation, the licensee for Vermont Yankee for not adequately protecting RCIC system cables (emergency power supply and control cables for the RCIC steam supply isolation valve V13-15) and ADS circuits (for all four ADS safety relief valves) from maloperation due to hot shorts from fire. In addition, for a fire in reactor building area RB-3, potential fire induced circuit failures associated with the wiring and terminals for ADS valve SRV-71A could have prevented the reactor vessel from being depressurized. These cited conditions of noncompliance resulted in the issuance of a Severity Level III Notice of Violation.

Region II

On May 6, 1994 Tennessee Valley Authority (TVA) reported that at Browns Ferry Nuclear Plant Unit 2 in portions of the reactor building and cable spreading room the power supply cables to two redundant (inboard and outboard) reactor water cleanup (RWCU) system primary containment isolation valves were not separated in accordance with the physical separation requirements of Section III.G.2 of Appendix R. In order to meet Appendix R post-fire safe shutdown requirements, one or both of these valves need to be shut to prevent the loss of reactor coolant inventory. In addition, TVA reported that associated circuits concerns existed when Unit 2 Raw Cooling Water (RCW) pump 1D (spare) was in use in support of Unit 2 operation (power circuit breaker closed). Due to lack of pump 1D power cable/electrical switchgear electrical separation and the existence of pump 1D power cable common enclosure (cable tray) concerns in multiple plant locations, a fire induced failure of the RCW pump 1D's trip circuit control cable, combined with a fault in the 1D power cable could cause a loss of power to the Unit ½ 4160-volt shutdown board "A" or the malfunction of a cable supplying power to other fire safe shutdown loads (refer to 50.73 Licensee Event Report 50-260/94-002).

On July 7, 1995, Florida Power Corporation notified the NRC that, as part of its re-evaluation of Crystal River Unit 3 Appendix R compliance, it had determined that circuits for the nuclear services closed cycle cooling water pumps (SWP) 1A and 1B terminated at the remote shutdown panel. Thus, for a fire in the remote shutdown panel area a fire could induce short circuits (hot shorts) or grounds and could fail both pumps. This would render the SW system unavailable (refer to 50.73 Licensee Event Report 50-302/95-12).

On May 25, 1997, Southern Nuclear Company, the licensee for the Farley Nuclear Plant reported that five Unit 2 motor operated valves had discrepancies in their control power wiring circuits that could potentially affect their ability to be remotely operated from the remote shutdown panel. The licensee identified that the control circuit for the PORV block valve was wired with the control power fuses for local and remote operation in series. In the event of a fire in the main control room, a fire-induced short could cause the failure of the remote shutdown fuses (refer to 50.72 Immediate Notification 32390).

Region III

On August 25, 1997, Illinois Power Company, the licensee for the Clinton Power Station, reported that the design of a control circuit fed from the 125 volt DC control power fuse for the Division 1 emergency diesel generator (EDG) feeder breaker was not in accordance with the plant's post-fire safe shutdown analysis (SSA). This circuit was routed through the main control room and did not incorporate isolation contacts at the remote shutdown panel transfer switch. The lack of isolation contacts would not ensure isolation of electrical faults when transfer/isolation of the breaker control circuit was attempted from the remote shutdown panel. Therefore, there was the potential that a fire in the main control room, with the loss of off-site power, could cause multiple DC ground faults and loss of control power from the Division 1 feed breaker (refer to 50.73 Licensee Event Report 50-461/97-021).

On October 22, 1996, the NRC cited Centerior Service Company, the licensee for the Davis Besse Nuclear Power Plant for the failure to protect control circuits for sixteen MOVs from fire-induced c uit failures which could result in spurious valve movement and secondary valve operator damage (IN 92-18). This cited condition of noncompliance resulted in the issuance of a Severity Level III Notice of Violation.

On January 12, 1998, Centerior Service Company reported that at Davis Besse Nuclear Power Station Unit 1, the tachometer circuit of the speed switch (speed sensing circuit) for the protected post-fire safe shutdown #1 emergency diesel generator (EDG) was not protected from the hot short effects of a fire in the control room or cable spreading room. The licensee reported that the EDG tachometer circuit lacked adequate protective fusing, and that circuit damage could occur before operators could manually isolate the circuit using a disconnect switch in the EDG room. Failure of the affected circuit could prevent automatic diesel start and prevent local operator control of the #1 diesel generator (refer to 50.73 Licensee Event Report 50-346/97-015).

On March 7, 1997 Detroit Edison Company discovered that the Fermi emergency equipment cooling water (EECW) system may not have been properly aligned for operation from the remote shutdown panel as specified by the shutdown procedure for a serious control room fire. The licensee determined that the EECW makeup valve from the EECW makeup tank could not open due to an electrical interlock with the reactor building closed cooling water to EECW isolation valves, unless the EECW valves were fully closed. There were no local transfer switches to override the interlock which prevents the EECW system from being aligned to its makeup tank from the Remote shutdown panel (refer to 50.72 Immediate Notification 31909).

On August 13, 1996 Consumers Power Company, the licensee for the Palisades Nuclear Plant was cited by the NRC for failing to promptly correct Appendix R related post-fire safe shutdown conditions which included (1) the failure to properly coordinate the potential transformer circuit primary and secondary fuses such that they protect Emergency Diesel Generator 1-1, (2) susceptibility of required post-fire safe shutdown motor operated valves to fire-induced spurious operations and secondary mechanical damage (IN 92-18), (3) lack of separation (with fire rated barriers) of redundant emergency diesel power and control circuits, and (4) failure to coordinate the main supply fuses for the 125 Volt dc panels with branch circuit breakers. These cited conditions of noncompliance resulted in the issuance of a Severity Level III Notice of Violation.

On October 30, 1997, Consumers Energy Company reported a Palisades Nuclear Plant safe shutdown analysis error with respect to the spurious opening of two main steam atmospheric dump valves (ASDVs). In order to terminate a steam generator blowdown, certain manual actions were required to be taken within 10 minutes. During a re-analysis it was determined that all four ADSVs and the turbine by-pass could spuriously open. After re-evaluation of this condition, local manual actions to close the ADSVs were determined to be required to be completed within 6 minutes (Refer to 50.73 Licensee Event Report 50-255/97-10).

On October 10, 1997, during its Appendix R Program Engineering Re-analysis, the Consumers Power Company reported that at its Palisades Nuclear Plant it failed to properly evaluate the potential for spurious opening of Service Water cross-tie valves. These three valves supply backup service water to cool engineered safeguards pump seals and bearings in the event the component cooling water system is lost. Specifically, a fire in one of several plant areas had the potential to spuriously open any one of these valves which would result in a loss of component cooling water inventory to the low pressure service water system (refer to 50.73 Licensee Event Report 50-255/97-008).

On September 12, 1997, Consumers Energy Company reported an Appendix R scenario at Palisades Nuclear Plant in which a single "hot short" could cause spurious operation of any one of three air operated valves. Such spurious operation could drain the component cooling water system, which is required to cool the seals of engineered safety feature pumps required for post-fire safe shutdown. The valve air supplies were isolated as an immediate compensatory measure (refer to 50.72 Immediate Notification 32919).

On June 30, 1997, at the Point Beach Nuclear Plant, Wisconsin Electric Power Company reported that it had identified two power distribution cables that provide power to units 1 and 2 essential buses needed for safe shutdown, located in the auxiliary feedwater pump room, which do not meet the fire protection and separation requirements of Section III.G of Appendix R to 10CFR Part 50 (refer to 50.72 Immediate Notification 32561).

On September 26, 1997, Commonwealth Edison Company (CECo) reported deficiencies with the post-fire safe shutdown procedures at Quad Cities Units 1 and 2. Because of these deficiencies and the lack of confidence in the Quad Cities associated circuit analysis (fuse breaker coordination of 480VAC, 250VDC, and 125VDC shutdown loads), CECo declared all Appendix R post-fire safe shutdown paths inoperable. In addition, as a result of this condition, the licensee shutdown Unit 2 (refer to 50.72 Immediate Notification 32994).

On February 27, 1997 Commonwealth Edison Company reported that on February 27, 1997, it discovered that the Quad Cities reactor water cleanup (RWCU) system had been identified as a high/low pressure interface requiring isolation during certain design basis fire. Failure to properly isolate the RWCU system could have prevented station operators from attaining safe shutdown prior to reactor level reaching top of active fuel. The potential blowdown path was administratively eliminated by isolating the power feed to the normally closed RWCU blowdown orifice bypass valve (refer to 50.73 Licensee Event Report 50-265/97-006).

Region IV

On July 25, 1996, Nebraska Public Power District reported that as part of a Cooper Nuclear Plant post-fire safe shutdown re-evaluation they had determined that certain assumptions made in their original post-fire safe shutdown and associated circuit analysis may not have been conservative. Specifically, for a postulated fire on elevation 931'-0" of the reactor building north which causes multiple cable ground faults, the reactor core isolation cooling system controller could become disabled (Refer to 50.72 Event Notification, number 30787). Prior to the licensee's identification of this condition, on April 17, 1996 the NRC cited Nebraska Public Power District for failure to protect control power circuitry for Diesel Generator 2 (the protected safe shutdown train power supply) from the effects of a postulated fire in the control room. A control room fire could result in the fire-induced failure of Diesel Generator No. 2 control power fuse prior to the manual isolation of the control circuit. This cited condition of noncompliance resulted in the issuance of a Severity Level III Notice of Violation.

On April 11, 1994, the NRC cited Entergy Operations, Inc., the Licensee for the River Bend Stations for failure to provide fire protection features for associated circuits that shared a common enclosure with alternative shutdown circuits, Standby Service Water Cooling Tower Fan circuits, a conduit containing Division III control power cables, control circuits for 4160 and 480 Volt circuit breakers supplying loads required for remote shutdown outside the main control room, and the reactor vessel level 8 trip circuitry and the breaker control circuitry for the reactor feedwater pumps. These cited conditions of noncompliance resulted in the issuance of a Severity Level III Notice of Violation.

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