RAS-E-126

July 18, 2008

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
ATOMIC SAFETY AND LICENSING BOARD

DOCKETED
US NRC
Rulemakings and
Adjudications Staff
Office of Secretary

Before Administrative Judges:
Lawrence G. McDade, Chairman
Dr. Richard E. Wardwell
Dr. Kaye D. Lathrop

In the Matter of	)	
	)	Docket Nos.
Entergy Nuclear Operations, Inc.	)	50-247-LR and
(Indian Point Nuclear Generating	)	50-286-LR
Units 2 and 3)	)	

SUPPLEMENTAL INTERVENOR PETITION BY WESTCHESTER CITIZEN'S AWARENESS (WESTCAN), ROCKLAND COUNTY CONSERVATION ASSOCIATION (RCCA), PUBLIC HEALTH AND SUSTAINABLE ENERGY (PHASE), SIERRA CLUB-ATLANTIC CHAPTER (SIERRA CLUB), and ASSEMBLYMAN RICHARD BRODKSY (BRODSKY)

TEMPLATE = SECY-037

DS 03

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BRIEF ISSUE STATEMENT

Petitioners, WestCAN, RCCA, PHASE, Sierra Club, and Brodsky (Collectively referred to as "Petitioners") submit this Petition to add a new contention alleging that Entergy Nuclear Operation, Inc's (hereinafter "Entergy" or "Licensee") License Renewal Application ("LRA") does not set forth an aging management plan or safeguards to prevent accidental emergency shutdowns triggered by mircowatt electronic devices, currently in use such as digital cameras, cell phones, blackberrys, pacemakers, hearing aids, ipods, etc, or for such electronic devices which will be developed over the next 20 years. Petitioners request a hearing on this issue in accordance with 10 C.F.R. §2.309.

On March 24, 2008, an unplanned shutdown caused by the electronics of a digital camera clearly identify an important factor in the effective and safe aging management. Entergy's LRA is silent on any aging management plans to deal with new technologies which can cause unplanned shutdowns and therefore the LRA is insufficient and incomplete.

TIMELINESS AND STANDARDS FOR REVIEW

Pursuant to 10 C.F.R. § 2.309(f)(2) contentions may be amended or new contentions filed after the initial filing only with leave of the presiding officer upon a showing that: (i) The information upon which the amended or new contention is based was not previously available; (ii) The information upon which the amended or new contention is based is materially different than information previously available; and (iii) The amended or new contention has been submitted in a timely fashion based on the availability of the subsequent information.

The proposed new contention satisfies the regulators requirement by providing a specific statement of the contention, an explanation of the basis, a demonstration that it is within scope, and a determination of material issues that are in dispute. Additionally, the new contention is timely because it is based on highly significant information that was not available at the time Petitioners filed their initial Petition on December 10, 2007. The new contention based is on the new information that was not known until June 12, 2008, when the incident was published in a news article stating that Indian Point had been shut down in March of 2008.

This contention meets the 6 part test for admissibility under 10 C.F.R. §2.309(f)(1) because: 1. It raises a substantial issue of fact that is in dispute and asserts a specific statement of issue of fact; 2. Petitioners have set forth above a brief explanation pursuant to Section 2.309(f)(1)(ii); 3. The contention is within scope pursuant to Section 2.309(f)(1)(iii) as it involves the safe shut down of the plant as part of aging management plan or lack thereof; 4. This contention raises a material issue pursuant to Section 2.309(f)(1)(iv); 5. The contention is supported by the facts pursuant to Section 2.309(f)(1)(v); 6. This contention raises a genuine dispute of material law or fact pursuant to Section 2.309(f)(1)(vi).

BRIEF STATEMENT OF FACTS

On June 12, 2008 Petitioners and the public first learned through a news article in the Journal News that "a digital camera accidentally triggered the emergency shutdown of the Indian Point 2 nuclear reactor in March 2008, according to documents obtained by The Journal News. The two-day work stoppage cost plant owner Entergy Nuclear about $2 million in revenue, based on industry estimates. Federal regulators said radio frequencies from a camera that was too close to a control panel interfered with the plant's main boiler feedwater pump, which provides water to four steam generators." (Attached hereto and made a part hereof as Exhibit 1 Journal News article by Greg Clary).

The incident report by the Nuclear Regulatory Commission ("NRC") dated March 24, 2008 does not indicate that the reason the reactor was manually tripped thereby causing a virtual shutdown at Indian Point facilities. (Attached hereto and made a part hereof Exhibit #2 Event Number 44089). The incident report states that:

> "The Indian Point Unit 2 reactor was manually tripped from 94% power at 2216 on 3/23/08 due to a loss of speed on 22 main boiler feed pump (MBFP). Indian Point Unit 2 is currently in mode 3 with all automatic actions for a manual reactor trip occurring as required. Indian Point Unit 2 was in a coast down in advance of a scheduled refueling outage."

The May 22, 2008 Licensee Event Report 2008-001-00, "Manual Reactor Trip Due to Decreasing Steam Generator Levels Caused by Loss of Feedwater Flow as a Result of a Feedwater Pump Speed Control Malfunction" (attached hereto and made a part hereof as Exhibit 3) states for the first time that:

> The Auxiliary Feedwater System automatically started as expected due to Steam Generator low level from shrink. The direct cause was radio frequency interference (RFI) from camera use near a MBFP speed control signal processer. The root cause of the event was lack of knowledge that a digital camera is an RFI source that can produce adverse effects on digital control components. Contributing causes were a poor choice of the controlling procedure for camera activity, poor change management in implementing the review requirements without providing adequate review tools, failure to follow procedure. Corrective actions include preparation of specific procedural guidance on electronic interference sources, creating a change management plan to track implementation of the new procedure and process, perform a needs analysis for training, and a site communication of this event and lessons learned from the event.

The report further states that:

> The direct cause of the malfunction of the MBFP Lovejoy control system was the RFI from an energized camera due to close proximity either from the camera digital circuitry itself or from the electrical discharge of a large capacitor through the xenon flash with the Lovejoy control system for MBFP speed control. The root cause was a lack of knowledge that a digital camera is a source of RFI which, when within a critical range, for critical digital equipment can cause adverse effects. The CR staff and Planner were not aware that just having a digital camera turned on in close proximity to other digital equipment could cause a problem. Although RFI is a known phenomenon with a potential for un-intentional effects on electronic equipment, digital photography as an RFI source was not recognized or understood.

The License Event Report ("LER") dated May 22, 2008 stated that this was the first indication the Licensee's awareness of a causal connection of the plant trip on March 23, 2008 and a camera being used in the control room. The apparent cause of the virtual shutdown was set aside when Indian Point facilities were restarted after the refueling outage and experienced a second shutdown in April of 2008. (Petitioners' question the wisdom of restarting the plant when the initiating event causing the unexplained loss of MBFP flow was yet not understood. This may be a violation of plant Technical Specifications).

Page 5 of the LER states that a "Past Similar Events review" was performed of the past three years for events that involved a RT from a malfunction of the MBFP speed control. There were no similar LER's identified.

The scope of the incident review was incorrect. Scope should have evaluated, analyzed and measured the effects of any radio frequency emitter on the plant's primary safety systems. Additionally, signal frequency, strength, and component susceptibility should have been researched for similar events.

The nuclear industry does not have guidance or rules in place to preclude these events from continuing to occur during the proposed new superseding license period. The event report incorrectly concluded that this event had no effect on the health and safety of the public. The conditions leading to the plant trip were unknown at the time of the event—and required extensive analysis to understand. It was only by chance, not by engineering design, that when the RT was initiated it did not lead to other transients or accidents.

The design and operational controls must be established to prevent similar types of incidents. In assigning limits to RFI frequency, strength, and physical location, or shielding vulnerable components required for safe operation the change process cannot simply allow an ex-post facto change into the facility. The designers never envisioned use of RFIs in camera equipment, or other microwatt devices affecting the design function or operation of the plant.

Today these devices are used frequently. Signal strength could disrupt safe operation of the facility by devices much farther than 18 to 24 inches from the control panel as were the facts in this event.

For there to be adequate aging management of the plant consideration must be given to sources of RFIs from multiple devices, from multiple locations. Forecasting and installing shielding for RFIs being designed in the coming decades is not unreasonable.

In fact the EPRI Portfolio Report 2009 regarding Instrumentation and Control Aging and Obsolescence (supplemental) states that:

> "Nuclear plants pursuing extended operation will inevitably replace much of their aging and obsolete instrument and control (I&C) systems. These replacements will likely be phased in over several years, demanding careful planning and maintenance practices for both existing and replacement systems. Further, the transition to digital technology presents

many new challenges for nuclear utilities and their traditional suppliers, and industry guidance is needed. For example, the digital systems being implemented now are expected to become obsolete far more rapidly than their analog predecessors, and this should be anticipated in the long-term planning." (Exhibit 4 EPRI Portfolio 2009 - 41.07.01.02 Instrumentation and Control Aging and Obsolescence (supplemental)).

In a report entitled Evaluating the Effects of Aging on Electronic Instrument and Control Circuit Boards and Components in Nuclear Power Plants, dated May 2005 for the U.S. Department of Energy, (attached hereto as Exhibit 5: Evaluating the Effects of Aging on Electronic Instrument and Control Circuit Boards and Components in Nuclear Power Plants Final Report), the report concluded that:

> The nuclear power industry is currently facing increasing obsolescence issues with original equipment installed for instrumentation, control, and safety system applications. These systems, frequently more than thirty years old, are experiencing aging-induced failures in electronic boards and components. These failures can cause plant trips and reduce the reliability and availability of systems. Most plants take a policy of running to failure and/or periodic replacement—frequently without a good technical basis. Both of these approaches can be very costly. The industry needs a better understanding of the aging mechanisms and observable precursors to failure along with more cost-effective aging inspection, mitigation, and other aging management technologies.

These industry and governmental reports confirm that the current electronic boards and components are becoming obsolete and that Entergy's LRA does not adequately present an aging management plan that will assure adequate protection of public health and safety.

Petitioners' assert that the unplanned shut down caused by a digital camera clearly demonstrates a problem in current operations, which will be carried over into the proposed new license period. It is an unacceptable risk to public health and safety for mircowatt technologies to cause unplanned shutdowns, and the corrective actions proposed in the LER are insufficient in addressing the issue.

Shielding or establishing regulatory limits for these emitters should be considered in light of the Licensee's LRA. This is a primary example of a change that not only did not received the NSHC analysis, but received no analysis, other than a modest review prepared under the associated LER.

Petitioners assert that in order to establish these changes to the CLB and the subset DB a "no significant hazards analysis consideration" is also required a prior to this issue being resolved satisfactorily and controls implemented for not just this particular unit but for any licensee's facility with this vulnerability.

Petitioner's assert that Entergy's LRA does not set forth an aging management plan or safeguards to prevent accidental emergency shutdown triggered by mircowatt electronic devices currently in use such as digital cameras, cell phones, pacemakers, hearing aids, etc. nor for such electronic devices which will be developed and used popularly over the next 20 years.

The March 2008 unplanned shutdown caused by the microwatt electronics of a digital camera clearly identify an important factor in the effective and safe aging management. The LRA is silent on any aging management plans to safeguard against new and existing microwatt technologies which can cause unplanned shutdowns requiring manual response and thereby failure to adequately protection the public health and safety.

CONCLUSION

Petitioners contend that the issue of mircowatt electronic equipment currently in common use causing unplanned shutdowns of Indian Point is an aging management concern. Mircowatt electronic equipment currently in use and which will be use in the 20 years of the proposed new superseding license period are not addressed in the LRA, and therefore the LRA does not present a comprehensive aging management plan or necessary safeguards to prevent unplanned shutdowns in order to adequately protect public health and safety, nor does it meet the requirements of 10 C.F.R. § 54. Therefore the NRC should deny the Entergy's LRA as being inadequate and incomplete, and grant Petitioners request for a hearing on this Contention.

Submitted by:

Susan Shapiro, Esq.

Representing: Westchester's Citizen's Awareness Network, Rockland County Conservation Association, Public Health and Sustainable Energy, Sierra Club- Atlantic Chapter, and Assemblyman Richard Brodsky

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket Nos. 50-247/286-LR
(Indian Point Nuclear Generating)	)	ASLBP No. 07-853-03-LR-BD01
Units 2 and 3)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing Petitioners' WestCAN et. al Petition for admissibility of a new contention has been served upon the following by electronically as shown to the address below, this 18th day of July, 2008.Hard copies have only been sent to the Office of the Secretary, as required by the February 1, 2008 ASLB Order.

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Atomic Safety and Licensing Board Panel

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Susan Shapiro, Esq.

* Original and two copies

EXHIBIT 1

JOURNAL NEWS

Camera caused reactor shutdown in March

Greg Clary The Journal News June 12, 2008

BUCHANAN - A digital camera accidentally triggered the emergency shutdown of the Indian Point 2 nuclear reactor in March, according to documents obtained by The Journal News.

The two-day work stoppage cost plant owner Entergy Nuclear about $2 million in revenue, based on industry estimates.

Federal regulators said radio frequencies from a camera that was too close to a control panel interfered with the plant's main boiler feedwater pump, which provides water to four steam generators.

Once the water levels dropped low enough because of the malfunction, workers were forced to shut down the plant two days earlier than planned for a scheduled refueling.

No radiation was released.

"After they did their trouble-shooting on this event, they determined that it was initiated by someone taking photos," said Neil Sheehan, a spokesman for the Nuclear Regulatory Commission. "The direct cause was radio frequency interference from the camera. All that had to happen was for the camera to be on."

Sheehan said the camera was in close proximity to the control panel, and that the company has changed its photography procedures because of the incident. Entergy researched the cause and reported it to the NRC as required.

The agency is confident that radio frequency interference with the control panel cannot be carried out from a remote distance, Sheehan said.
Other camera events have occurred in control rooms over the years, including two camera flashes in 1997 that caused the release of halon gas at the Haddam Neck Plant in Connecticut.

The gas was used for fire suppression, and the automatic sensors activated with the sequence of flashes. The plant was preparing for decommissioning, but the control room had to be evacuated for an hour.

EXHIBIT 1 - 2

In a 1994 incident that has become legend in the industry, the changing of a light bulb caused an emergency shutdown at the Limerick nuclear plant in Montgomery County, Pa.

A 10-watt bulb broke during replacement, causing a short circuit and tripping generator cooling pumps and then pumps that cooled the reactor. Quickly, workers had no choice but to shut down.

Robyn Bentley, a spokeswoman for Entergy Nuclear, said the Indian Point incident showed the need to have employees and visitors shut off digital devices such as BlackBerries.

Entergy has since changed its procedures related to camera use in control rooms, and NRC officials said they were passing the information on the incident to the other 103 nuclear plants in the country.

Bentley said the company was taking photographs as part of its planning for the upcoming outage, which was to start March 25. The unplanned shutdown occurred March 23.

A subsequent unplanned shutdown April 22 as the nuclear plant in Buchanan was restarting has put Indian Point 2 on the brink of a lower safety rating - and increased scrutiny from federal regulators.

One more unplanned shutdown within a 12-month rolling period would change Indian Point 2's performance rating from the top level of green to the second level of white.

EXHIBIT 2

Power Reactor

Facility: INDIAN POINT
Region: 1 State: NY
Unit: [2] [ ] [ ]
RX Type: [2] W-4-LP,[3] W-4-LP
NRC Notified By: JOHN DIGNAM
HQ OPS Officer: JOHN KNOKE

Emergency Class: NON EMERGENCY
10 CFR Section:
50.72(b)(2)(iv)(B) - RPS ACTUATION - CRITICAL
50.72(b)(3)(iv)(A) - VALID SPECIF SYS ACTUATION

Event Number: 44089

Notification Date: 03/24/2008
Notification Time: 01:56 [ET]
Event Date: 03/23/2008
Event Time: 22:16 [EDT]
Last Update Date: 03/24/2008

Person (Organization):
NEIL PERRY (R1)

Unit	SCRAM Code	RX CRIT	Initial PWR	Initial RX Mode	Current PWR	Current RX Mode
2	M/R	Y	94	Power Operation	0	Hot Standby

Event Text

MANUAL REACTOR TRIP - LOSS OF SPEED ON MBFP

"The Indian Point Unit 2 reactor was manually tripped from 94% power at 2216 on 3/23/08 due to a loss of speed on 22 main boiler feed pump (MBFP). Indian Point Unit 2 is currently in mode 3 with all automatic actions for a manual reactor trip occurring as required. Indian Point Unit 2 was in a coast down in advance of a scheduled refueling outage.

"The reactor was manually tripped as required by abnormal operating procedure 2-AOP-FW-001. All control rods inserted on the trip. No safety or relief valves lifted due to the trip. The motor driven aux feedwater pumps automatically started on low steam generator level and are being used to maintain steam generator level. Condenser steam dumps are maintaining reactor temperature. The Unit 2 electrical lineup is the normal shutdown electrical lineup. The licensee has notified the state Public Service Commission.

"Unit 3 was unaffected and remains in mode 1 at 100% power."

The licensee has notified the NRC Resident Inspector. The licensee expects to issue a media release.

EXHIBIT 3 - 1

Entergy

Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, N.Y.. 10511-0249
Tel (914) 734-6700

J. E. Pollock
Site Vice President
Administration

May 22, 2008
Indian Point Unit No. 2
Docket No. 50-247
NL-08-075

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, D.C. 20555-0001

Subject: Licensee Event Report # 2008-001-00, "Manual Reactor Trip Due to Decreasing Steam Generator Levels Caused by Loss of Feedwater Flow as a Result of a Feedwater Pump Speed Control Malfunction"

Dear Sir or Madam:

Pursuant to 10 CFR 50.73(a) (1), Entergy Nuclear Operations Inc. (ENO) hereby provides Licensee Event Report (LER) 2008-001-00. The enclosed LER identifies an event where the reactor was manually tripped, which is reportable under 10 CFR 50.73(a)(2)(iv)(A) . As a result of the reactor trip, the Auxiliary Feedwater System was actuated which is a system listed in 10 CFR 50.73(a)(2)(iv)(B) that is reportable under 10 CFR 50.73(a)(2)(iv)(A). This condition has been recorded in the Entergy Corrective Action Program as Condition Report CR-IP2-2008-01333.

There are no new commitments identified in this letter. Should you have any questions regarding this submittal, please contact Mr. Robert Walpole, Manager, IPEC Licensing at (914) 734-6710.

Sincerely,

J. E. Pollock
Site Vice President
Indian Point Energy Center

cc: Mr. Samuel J Collins, Regional Administrator, NRC Region I
NRC Resident Inspector's Office, Indian Point 2
Mr. Paul Eddy, New York State Public Service Commission
INPO Record Center

IE22
NRR

EXHIBIT 3 - 2

NRC FORM 366 (6-2007) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
INDIAN POINT 2	05000-247	1 OF 4

4. TITLE Manual Reactor Trip Due to Decreasing Steam Generator Levels Caused by Loss of Feedwater Flow as a Result of Feedwater Pump Speed Control Malfunction

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV. NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
3	23	2008	2008-	001 -	00	05	22	2008		05000
									FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 94.5%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER *(Include Area Code)*
Lizabeth Lee, Supervising Engineer	(914) 827-7636

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	JK	PMC	L253	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO				

16. ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced type written lines)*

On March 23, 2008, at 2216, with reactor power at 94.5%, as part of a planned coast down for a scheduled refueling outage, a manual reactor trip (RT) was initiated as a result of 22 Main Boiler Feed Pump (MBFP) rapid speed reduction causing lowering steam generator levels. All control rods inserted and all required safety systems functioned properly. The plant was stabilized in hot standby with decay heat being removed by the main condenser. There was no radiation release. The Emergency Diesel Generators did not start as off-site power remained available. The Auxiliary Feedwater System automatically started as expected due to Steam Generator low level from shrink. The direct cause was radio frequency interference (RFI) from camera use near a MBFP speed control signal processer. The root cause of the event was lack of knowledge that a digital camera is an RFI source that can produce adverse effects on digital control components. Contributing causes were a poor choice of the controlling procedure for camera activity, poor change management in implementing the review requirements without providing adequate review tools, failure to follow procedure. Corrective actions include preparation of specific procedural guidance on electronic interference sources, creating a change management plan to track implementation of the new procedure and process, perform a needs analysis for training, and a site communication of this event and lessons learned from the event. The event had no effect on the public health and safety.

EXHIBIT 3-3

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Indian Point Unit 2	05000-247	2008	- 01	- 00	2 OF 4

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Note: The Energy Industry Identification System Codes are identified within the brackets {}.

DESCRIPTION OF EVENT

On March 23, 2008 at 2216 hrs, while at approximately 94.5% reactor power, as part of a planned coast down for a scheduled refueling outage, the 22 Main Boiler Feed Pump (MBFP) {SJ} speed rapidly decreased to 2400 RPM. The speed reduction resulted in reduced feedwater flow to steam generators (SG) {AB} lowering SG water levels. Automatic Turbine runback initiated due to the speed reduction of 22 MBFP below 3300 RPM. Due to unloading of the 22 MBFP, Control Room (CR) {NA} Operators performed immediate actions of Feedwater Abnormal Operating Procedure {2-AOP-FW-1) and manually initiated a Reactor Trip (RT) {JC}. All control rods {AA} fully inserted and all safety systems responded as expected. The event was recorded in the Indian Point Corrective Action Program (CAP) as condition Report CR-IP2-2008-01333. Immediately prior to the event a planner was photographing the MBFP speed control {JK} Lovejoy signal processor power supplies {JX} for an upcoming refueling outage. The event occurred when the planner had taken a fourth picture at approximately 18 to 24 inches from the equipment. System Engineering (SE) reviewed plant data and verified that at the time of camera use steam flow to 22 MBFP turbine decreased very rapidly and 22 MBFP suction flow decreased to approximately 2800 GPM. The Lovejoy signal processor contains both analog and digital components with the digital regulation board (governor board) {90} the main digital component. Engineering concluded that the cause of the malfunction of the 22 MBFP Lovejoy controls was the camera based on discussion with Lovejoy manufacturer and review of the camera specifications. The camera is rated by the Federal Communications Commission (FCC) as radio frequency interference (RFI) device. The CR staff and planner were not aware that just having a digital camera turned on in close proximity to other digital equipment could cause a problem. The planner had been taking pictures with the same camera in CR for the last several months for outage preparation. The CR staff was used to having staff members including engineers, simulator staff and Public Relations personnel taking photographs in the CR. The planner request to CR staff to photograph "power supplies" was allowed since previous activities had no adverse effect. Planning personnel did not perceive a correlation between the use of a camera and its effect on plant equipment but were aware of Security requirements for camera use on site. The procedure for camera use on site is security procedure EN-NS-214,"Camera Controls for Access and Use." The responsibility of SE performing a technical evaluation was not listed under section "5.0 Responsibilities" of procedure EN-NS-214, and the Indian Point Energy Center Nuclear Management Manual (NMM) Review and Approval Form of new procedure EN-AD-101,"Change Management," did not indicate any requirement for cross-discipline review. Based on interviews, no communication plan was used for the new procedure.

EXHIBIT 3-4

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

Interviews with plant staff revealed that none of the SEs interviewed were aware that the security passes for cameras required permission from SE for flash photography inside electronic cabinets or computers. All personnel interviewed stated they knew to carry their camera passes when using the cameras but thought the requirements were security-related issues only. They were also not aware that digital cameras were RFI emitters but were aware of flash photography effects on EPROM. Personnel lack of knowledge, coupled with previous use of photography in the CR without any adverse issues along with camera passes viewed as permits to carry and use the camera for security reasons only, presented a human performance error trap.

Cause of Event

The direct cause of the malfunction of the MBFP Lovejoy control system was the RFI from an energized camera due to close proximity either from the camera digital circuitry itself or from the electrical discharge of a large capacitor through the xenon flash tube, which interfered with the Lovejoy control system for MBFP speed control. The root cause was a lack of knowledge that a digital camera is a source of RFI which, when within a critical range, for critical digital equipment can cause adverse effects. The CR staff and Planner were not aware that just having a digital camera turned on in close proximity to other digital equipment could cause a problem. Although RFI is a known phenomenon with a potential for un-intentional effects on electronic equipment, digital photography as an RFI source was not recognized or understood.

The following Contributing Causes (CC) were identified: CC1: Poor choice of placing technical requirements in a security procedure caused a mindset that the only restrictions related to the use of cameras are related to security issues, CC2: Poor change management. Neither the Staff obtaining camera passes or SE who was supposed to evaluate use of cameras around sensitive electronic equipment were aware of the requirement, CC3: Failure to follow procedure. The planner failed to follow procedure EN-NS-214 and the camera pass, and request permission from SE prior to camera use around electronic equipment.

Corrective Actions

The following corrective actions have been or will be performed under Entergy's Corrective Action Program to address the cause and prevent recurrence:

- The planning staff were coached on the requirements of EN-NS-214 and management's expectations on camera use in posted areas and referenced the guidance in procedure IP-SMM-MA-102, "Site communications."
- Engineering staff were briefed of the event and lessons learned and requirements of EN-NS-214 for camera use inside electronic cabinets or computers and referenced the guidance in procedure SMM-MA-102.
- The plant staff were notified of the event and lessons learned and requirements of EN-NS-214 for camera use inside electronic cabinets or computers and referenced the guidance in procedure SMM-MA-102.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Indian Point Unit 2	05000-247	2008	- 01 -	00	4 OF 4

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

- A Shift Order was issued to operations providing the event, lessons learned, and the direction to prohibit the use of all cameras or any known RF emitter in the CR and referenced the guidance in procedure SMM-MA-102.
- Develop guidance/fleet procedure on electronic interference sources and their control and use in sensitive areas. Scheduled completion is July 31, 2008.
- Evaluate this CR topic and related information for both initial orientation and General Employee continuing training. Issue any tracking/implementation actions resulting from the needs analysis process. Scheduled completion is July 31, 2008.
- Develop a change management plan related to electronic interference based on digital equipment. Scheduled completion is July 31, 2008.

Event Analysis

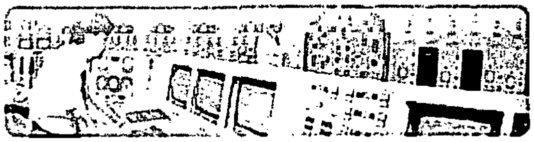
The event is reportable under 10CFR50.73(a)(2)(iv)(A). The licensee shall report any event or condition that resulted in manual or automatic actuation of any of the systems listed under 10CFR50.73(a)(2)(iv)(B). Systems to which the requirements of 10CFR50.73(a)(2)(iv)(A) apply for this event include the Reactor Protection System (RPS) including RT and AFWS actuation. This event meets the reporting criteria because a manual RT was initiated at 22:16 hours, on March 23, 2008, and the AFWS actuated as a result of the RT. The malfunction of the MBFP speed control did not result in the loss of any safety function. Therefore, there was no safety system functional failure reportable under 10CFR50.73(a)(2)(v).

PAST SIMILAR EVENTS

A review was performed of the past three years of Licensee Event Reports (LERs) for events that involved a RT from a malfunction of the MBFP speed control. There were no similar LERs identified.

Safety Significance

This event had no effect on the health and safety of the public. There were no actual safety consequences for the event because the event was an uncomplicated RT with no other transients or accidents. Required primary safety systems performed as designed when the RT was initiated. There were no TS related components out of service or off normal status of any safety systems at the time of the RT. The AFWS actuation was expected as a result of low SG water level due to SG void fraction (shrink), which occurs after automatic RT from full load. There were no significant potential safety consequences of this event under reasonable and credible alternative conditions. The reduced FW flow for this event was bounded by the analysis in FSAR Section 14.1.9, "Loss of Normal FW." The AFWS actuated and provided required FW flow to the SGs. Main FW remained available. RCS pressure remained below the set point for pressurizer PORV or code safety valve operation and above the set point for automatic safety injection actuation. Following the RT, the plant was stabilized in hot standby.



Instrumentation and Control

Overview Supplemental Projects

Associated with this program are the following research projects. Click on a title below to view the project description in the box below.

- Instrumentation and Control Aging and Obsolescence (base)
- Instrumentation and Control Aging and Obsolescence (supplemental)
- Instrumentation and Control Improvements to Increase Reliability and Capacity and Reduce Costs (base)
- Instrumentation and Control Improvements to Increase Reliability and Capacity and Reduce Costs (supplemental)
- Regulatory and Technical Issues for Instrumentation and Control

Instrumentation and Control Aging and Obsolescence (supplemental)

Add to Interest Form

Issue

Nuclear plants pursuing extended operation will inevitably replace much of their aging and obsolete instrument and control (I&C) systems. These replacements will likely be phased in over several years, demanding careful planning and maintenance practices for both existing and replacement systems. Further, the transition to digital technology presents many new challenges for nuclear utilities and their traditional suppliers, and industry guidance is needed. For example, the digital systems being implemented now are expected to become obsolete far more rapidly than their analog predecessors, and this should be anticipated in the long-term planning.

Description

This project focuses on key issues impacting the technical and financial success of maintaining existing I&C systems and the transition to updated I&C systems. Research addresses aging and obsolescence and ways to more effectively replace systems and equipment when it is no longer feasible to cope with problems with existing systems and equipment. Current activities include developing guidance and technology transfer mechanisms based on lessons learned to help eliminate inadvertent plant trips and other undesired events caused by costly "learning curve" errors made with digital upgrades. The project will develop a set of expert guides to capture knowledge related to I&C topics so that knowledge will be available to less experienced staff even after resident experts are no longer available. Generic requirements also are being developed for safety system digital upgrades to help utilities develop bid specifications and evaluate proposed new systems. Additional supplemental funded opportunities will be identified in this area to address plant needs.

Value

- Avoid problems made in the past and avoid associated costs:
  - Inadvertent plant trips caused by unanticipated and undesired behaviors of new digital I&C systems
  - Design system problems discovered during installation and testing, when costs to correct are high
  - Life-cycle maintenance process issues after installation with high costs and limited redress
- Develop generic requirements for digital reactor protection systems/engineered safety features actuation systems to help plants prepare plant-specific requirements and bid specifications with greater confidence and with less time and cost to address aging and obsolescence

How to Apply Results

Members apply lessons learned to refine internal procedures for plant modifications and ensure that potentially problematic issues are specifically addressed at the earliest possible point in the plant modification process and tracked through implementation. Members can use generic conceptual designs to inform system upgrades, including realistic performance requirements and vendor evaluations.

Current Deliverables
Future Deliverables - There are no deliverables defined at this time.

yes

Exhibit 5 -1

Evaluating the Effects of Aging on Electronic Instrument and Control Circuit Boards and Components in Nuclear Power Plants

Technical Report

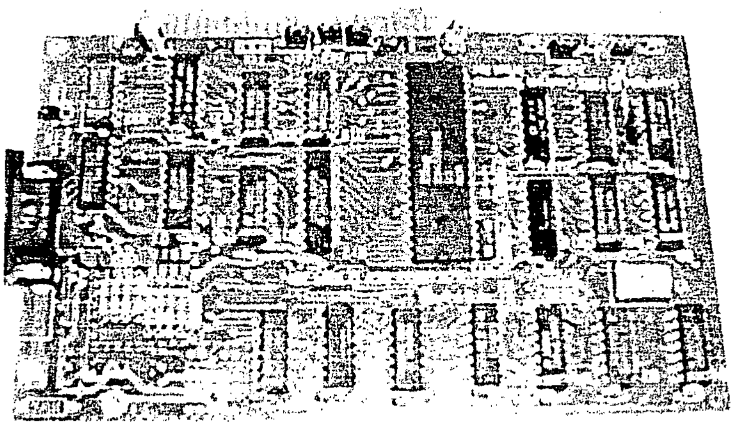


 EXHIBIT 5-2

Evaluating the Effects of Aging on Electronic Instrument and Control Circuit Boards and Components in Nuclear Power Plants

Final Report, May 2005

U.S. Department of Energy
1000 Independence Avenue
Washington, DC

EPRI Project Manager
J. Naser

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CITATIONS

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This report describes research sponsored by the Electric Power Research Institute (EPRI) and the U.S. Department of Energy (Award No. DE-FC07-03ID14536. Task #6).

The report is a corporate document that should be cited in the literature in the following manner:

Evaluating the Effects of Aging on Electronic Instrument and Control Circuit Boards and Components in Nuclear Power Plants. EPRI, Palo Alto, CA. and U.S. Department of Energy, Washington, DC: 2005. 1011709.

EXHIBIT 5-3

REPORT SUMMARY

Circuit boards used in the electronic instrument and control (I&C) systems of nuclear power plants may suffer from aging failures that can cause a plant trip or unavailability of plant systems. The overall objective of this study was to determine how precursors of failures in I&C circuit boards can be measured and how these measures can be used to estimate the probability of failure during the next operational period within a statistical confidence level. The study provides a framework for the identification of techniques that can be used to monitor circuit board component aging failure modes that could lead to a failure of the circuit.

Background

The nuclear power industry is currently facing increasing obsolescence issues with original equipment installed for instrumentation, control, and safety system applications. These systems, frequently more than thirty years old, are experiencing aging-induced failures in electronic boards and components. These failures can cause plant trips and reduce the reliability and availability of systems. Most plants take a policy of running to failure and/or periodic replacement—frequently without a good technical basis. Both of these approaches can be very costly. The industry needs a better understanding of the aging mechanisms and observable precursors to failure along with more cost-effective aging inspection, mitigation, and other aging management technologies.

Objectives

- To classify the failures of electronic boards and components used in I&C systems according to the type of measurable failure mode conditions
- To apply reliability criteria to the components to understand the likelihood of failure using existing data sources
- To propose potential measurement tools and models of failure for input into a condition monitoring and operational assessment process for predicting I&C board failures.

Approach

The project team reviewed the failure modes identified in EPRI reports 1003568 "Collected Field Data on Electronic Part Failures and Aging in Nuclear Power Plant I&C Systems" and 1008166 "Guidelines for the Monitoring of I&C Electronic Components," as supplemented by

EXHIBIT 5-4

technical papers, IEEE reliability meetings, and contacts with utility people. The team used data on failure modes from Military Handbook 217F to define the likelihood of failure for I&C electronic components. The team defined monitoring methods and techniques based on currently used methods and methods that have been used in laboratories for circuit board testing, for computer hardware testing, software verification and database integrity assurance.

Results

The report describes potentially useful techniques for monitoring the aging of I&C boards. The techniques have been grouped into six methods: periodic testing, reliability modeling, resistance measures, signal comparison, external (passive) measures, and internal (active) measures, each representing distinct theoretical approaches to detection and evaluation. Each technique has significant advantages and disadvantages. The design of hardware and software monitoring systems increases in complexity as the methods become more precise in their ability to measure aging factors, but the technical tools that can be applied to monitoring within the methods have also clearly improved within the last few years as computers and networks have been enhanced to rapidly process large amounts of data.

The report provides a decision process for selecting those circuits and components that could benefit from an upgraded approach for monitoring the effects of aging and highlights areas where future R&D is needed to establish firm recommendations for I&C systems. The report also assesses the relative costs and technical benefits of upgrading circuit-monitoring systems.

EPRI Perspective

As the nuclear power industry is facing increasing aging and obsolescence issues, one area that needs attention is the aging of electronic boards and components used in I&C systems. Existing methods of functional testing of I&C systems typically detect circuit failures after they occur whereas the new monitoring techniques provide indications of failure while the circuit is still functional. This information will make it possible to maximize the operating life of components without suffering circuit failure.

This report presents a number of specific techniques for improving the ability to monitor aging induced changes in circuits and board components that could lead to board failure. Some promising techniques are discussed that have been used in applications outside of electronic circuit board monitoring. Additional R&D efforts are needed to test, confirm, and demonstrate the viability of circuit board monitoring techniques for use as a predictive tool to detect aging induced changes that can lead to circuit failure. Additional engineering studies need to be completed to better quantify the implementation and operational costs and benefits of the viable techniques and to provide sufficient justification for their implementation.

Keywords

Instrumentation and control systems
Electronic boards and components
Electronic board aging
Electronic components aging
Circuit boards
Aging management
Reliability
Predictive maintenance

EX#11BIT 5 - 5

ABSTRACT

This report addresses understanding the effects of aging on electronic instrument and control (I&C) circuit boards in nuclear power plants. The issue is that circuit boards used in I&C systems may suffer from aging failures that can cause a plant trip or unavailability of plant systems. *The overall objective is to determine how precursors of failures in I&C circuit boards* can be measured, and how these measures can be used to estimate the probability of failure during the next operational period within a statistical confidence level. This initial study provides a framework for identification of techniques that can be used to monitor circuit board component aging failure modes that could lead to a failure of the circuit. This study has three tasks:

1. Provide (1) a review of information on circuit board aging failure descriptions, and (2) an identification of reliability data for I&C component failures.
2. Propose and review six uniquely different methods and associated techniques that could be applied to measure and predict the effects of aging within I&C boards and circuits.
3. Provide a systematic framework for deciding how to select circuit boards as well as methods for improving the process for monitoring aging effects of circuit boards in nuclear power plants - the framework includes a relative cost benefit assessment for each technique.

Many causes of I&C circuit board failure progress slowly. This opens the possibility for measuring the impacts of aging progression prior to complete failure. Measures of changes in electrical characteristics provide a basis for estimating the probability of failure during the next operational period. Simulation of the aging process can be used to produce a statistical confidence in the probability estimate. Such information can be used to support optimized maintenance planning and decisions.

The ideal result of this project is to define a framework for selecting techniques for aging management that can be applied to any circuit. The techniques should be easy to use and account for various modes of circuit component aging. This is an initial study to examine a wide range of issues and approaches for aging management in electronic systems. The existing state of the I&C systems is that most nuclear power plants were designed using analog control circuits and relay controlled safety circuits. Hardwired electric relays have become obsolete, as electronic circuits rely on integrated circuits and software controls to accomplish the same functions. Current technology permits software logic to replace relay logic and analog controls to activate safety and control circuits. Therefore, circuit boards can become obsolete in less than a decade and as this older equipment fails, the spare parts inventories become depleted and failures can't be easily repaired. Then there is an increasing need for older technology I&C systems to be upgraded or replaced. Even new solid state, integrated circuits and systems can become obsolete in a few years. Older nuclear power plants are in various stages of identifying

EXHIBIT 5 - 6

obsolescence, upgrading and replacing I&C systems. As a result the key control and protection systems in nuclear plants use a range of technologies in different plants, and even within the same plant. Thus, the state of I&C technology within plants may be considered as fragmented with respect to replacement components and the amount of digital versus analog I&C systems.

In an absolute sense the rate of failure and replacement for electronic circuit boards may not be considered high. However, from a regulatory point of view, and also relative to other major plant systems, the I&C system repair and replacement rates are relatively high. For example, individual circuit boards are typically repaired or replaced several times during the life of a plant. Therefore, this higher rate of circuit board replacements makes them of low concern as an aging issue in plant license extension, whereas, insulation on the wires connecting the I&C systems is a required aging issue that must be addressed in license extension applications.

Reviewing the descriptions of aging failures in circuit boards provides some very valuable insights. For example, a conclusion from review of information from EPRI (EPRI 2002, EPRI 2003 and EPRI 2004) is that the failure end state of most electronic components is either an open or short circuit. These findings help simplify the design of potential measurement systems by permitting the monitoring of each board circuit to be treated as an equivalent circuit with measurable electronic parameters such as voltage, impedance, resistance, current, and ground resistance. Changes in these parameters become precursor indications of degradation that could lead to a complete failure.

Aging induced failures (due to temperature, operating stress, quality of components, corrosion, and environment) are slow and many intermediate states of partial failure exist. Changes in the electrical parameter signals from the circuit can be measured before an inoperable condition is reached. In the case of a rapid shock induced failure (e.g., high voltage spikes, rapid corrosion or high temperature effects from fire, etc.) the time between the triggering condition and the component failure would be too short for corrective action to be taken before a failure.

A variety of technology and software methods can be used to develop improved monitoring, including continuous circuit monitoring, and active as well as passive testing approaches.

1. The EPRI reports developed by EdF (EPRI 2002 and EPRI 2004) show the impact of component failures involving capacitance and inductance that are sensitive to frequency variation tests.
2. Simple voltage tests could easily identify circuits that are drifting toward shorts or open circuits.

Replacement of an aging component with a spare board may not always be possible, because of obsolescence. If this is the case, then the ability to locate the failed component supports the process of replacing individual components on a board when replacement circuit boards are not available. This circuit board reconditioning can be enhanced by early identification of precursor failures.

Considerable failure data are available for electronic components. These data are complied as failures per unit time for like components. The grouping process reduces the details of the failure modes in MIL-HBK-217F (i.e., descriptions of the failure modes typically found in event data and root cause analysis are subsumed into the term "failure" so that the detail is lost). The MIL-HBK-217F data can be assigned with some judgment to the individual board components. Since a board or circuit contains a combination of many components, board failure rates can be approximated by a sum of failure rates for the components following the recommended

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July 18, 2008 (10:06am)

OFFICE OF SECRETARY
RULEMAKINGS AND
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July 19, 2008

Office of the Secretary
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Rockville, Maryland, 20852

RE: PETITION and EXHIBITS FOR SUPPLEMENTAL INTERVENOR PETITION BY WESTCHESTER CITIZEN'S AWARENESS (WESTCAN), ROCKLAND COUNTY CONSERVATION ASSOCIATION (RCCA), PUBLIC HEALTH AND SUSTAINABLE ENERGY (PHASE), SIERRA CLUB-ATLANTIC CHAPTER (SIERRA CLUB), and ASSEMBLYMAN RICHARD BRODKSY (BRODSKY)

To Whom It May Concern:

Please find enclosed one original and two copies of the Petition and Exhibits for the Supplemental Intervenor Petition. Inadvertently a set of Petitions without the exhibits was sent to you yesterday, therefore please only use this enclosed set with Exhibits as the filed version.

Sincerely yours,

Susan Shapiro

Cc: Sherwin Turk, Counsel for the NRC Staff
Kathryn M. Sutton, Esq.
Paul Bessette, Esq.
Martin O'Neill, Esq.