July 12, 2001

MEMORANDUM TO: Maggalean W. Weston, Senior Staff Engineer
ACRS

FROM: John D. Sieber, Chairman
Plant Operations Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON PLANT OPERATIONS, May 9, 2001, ROCKVILLE, MD

I hereby certify that, to the best of my knowledge and belief, the minutes of the meeting on the Reactor Oversight Process issued July 9, 2001, are an accurate record of the proceedings for that meeting.

John D. Sieber 7/12/01

John D. Sieber, Chairman Date

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 16, 2001

MEMORANDUM TO: ACRS Members

FROM: Maggalean W. Weston, Senior Staff Engineer
ACRS/ACNW

SUBJECT: CERTIFICATION OF THE SUMMARY/MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON PLANT OPERATIONS, MAY 9, 2001, ROCKVILLE, MD

The minutes of the subject meeting, issued July 12, 2001, have been certified as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc via Email: J. Larkins
J. Lyons
H. Larson
ACRS Staff Engineers
ACRS Fellows

Certified: J. Sieber
July 12, 2001

CERTIFIED

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
PLANT OPERATIONS SUBCOMMITTEE
REACTOR OVERSIGHT PROCESS
May 9, 2001
ROCKVILLE, MARYLAND

The ACRS subcommittee on Plant Operations held a meeting on May 9, 2001, with representatives of the NRC staff. The purpose of this meeting was to discuss the status of the Reactor Oversight Process (ROP). The meeting was open to the public. Mrs. Maggalean W. Weston was the cognizant ACRS staff engineer and Designated Federal Official (DFO) for this meeting. There were no written comments provided by the public. The meeting was convened by the Subcommittee Chairman at 8:30 a.m. May 9, 2001, and adjourned at 2:40 p.m. that day.

Attendees

ACRS Members/Staff:

J. Sieber, Chairman	P. Ford, Member	W. Shack, Member
G. Apostolakis, Member	T. Kress, Member	R. Uhrig, Member
M. Bonaca, Member	G. Leitch, Member	M. W. Weston, DFO

NRC Staff:

D. Hickman, NRR	Alan Madison, NRR	P. Koltay, NRR
D. Coe, NRR	M. Johnson, NRR	J. Hannon, NRR
T. Frye, NRR	W. Dean, NRR	D. Allsopp, NRR
P. Wilson, NRR	S-M. Wong, NRR	J. Hyslop, NRR
E. Cobey, NRR	J. Thompson, NRR	M. Sartorius, NRR
M. Salley, NRR	S. Stein, NRR	T. Boyce, NRR
A. El-Bassioni, NRR	J. Jacobson, NRR	G. Parry, NRR
Leon Whitney, NRR	P. Qualls, NRR	

There were no members of the public in attendance during this meeting. No members of the public participated in the meeting discussions.

The presentation slides and handouts used during the meeting are attached to the Office Copy of the Minutes. The presentation to the subcommittee is summarized below.

CHAIRMAN'S COMMENTS

J. Sieber, Subcommittee Chairman, convened the meeting. Mr. Sieber stated that the purpose of the meeting was to discuss the Reactor Oversight Process, specifically, the initial implementation status, the significance determination process, performance indicators, and some cross-cutting issues. The subcommittee chairman asked for comments from other ACRS committee members.

NRR Staff Presentation

Mr. Michael Johnson of the Inspection Program Branch introduced the presenters. He indicated that staff from the Plant Systems Branch would also be talking about specific topics.

REACTOR OVERSIGHT PROCESS (ROP)

Initial Implementation Status

Mr. Johnson stated that the first year of initial implementation which began April 2, 2000, is ending. Regions are conducting their end-of-cycle reviews in order to provide the results of the ROP for the year to the licensees and other external stakeholders. Following the end of cycle, there will be an Agency Action Review Meeting (AARM) to discuss the results of the ROP. The ROP process has been substantially exercised during this first year of implementation and several significant changes were made based on lessons learned that needed to be corrected. The process is more objective, more understandable, and more predictable. This conclusion is based on data collected and feedback from stakeholders. Overall, the staff is satisfied with the initial implementation of the ROP.

Dr. Apostolakis stated that the ultimate goal of the ROP was to make decisions. He asked if any decisions had been made using this process that would have been different if the old process had been followed. Mr. Johnson stated that he believed so, even though they are still evaluating the matrix.

Significance Determination Process (SDP)

The inputs from the PIs and from the inspections are applied to the thresholds to decide whether increased regulatory action should be taken in accordance with the Action Matrix. The Action Matrix is a structured matrix that enables the staff to determine what the agency responses should be based on plant performance. When inspections are done, the SDP provides an objective way to view the findings from inspections to determine whether the findings are significant and whether they warrant taking increased regulatory action. In other words, the matrix provides a methodology for equating inspection findings and PIs. The SDP is necessary to characterize the significance of inspection findings as one of the two inputs to the Action Matrix. The other input is the PIs.

During the presentation, four examples of SDPs were given for reactor safety. Examples of inspection findings for no-color, green, and non-green were presented. The no-color findings category is a finding that does not affect a cornerstone, or that does not have extenuating circumstances and cannot be processed by an SDP. A no-color inspection finding as defined in Inspection Manual Chapter (IMC 0610) is one that potentially impacts NRC's ability to perform its regulatory function. These findings are treated as non-cited violations consistent with the NRC Enforcement Policy and confirmed as an entry into the licensee's corrective action program.

A Green inspection finding as defined in IMC 0610 affects a cornerstone, e.g., affects a safety function or operability. These findings are treated as Phase 1 SDPs because system function is affected, but operability and safety function are maintained. These are also confirmed as an entry into the licensee correction action program.

A Non-green inspection finding (White, Yellow, Red) as defined in IMC -610 is a finding that significantly affects a cornerstone . These findings are treated as Phases 2 or 3 SDPs because they represent a loss of safety function or operability and are, therefore, in a regulatory response band.

Another SDP example was given on fire protection. It was stated that this SDP uses PRA techniques and data generally accepted by the fire risk community, but is still an evolving process. The presentation included identification of the findings, the configuration of the room in which the findings existed, and the basis for the degradation levels and failure probabilities

Performance Indicators (PIs)

The PIs use objective data to monitor performance in each of the cornerstones, and are built around changes in core damage frequency. The process of developing thresholds for the ROP showed that a licensee response band was needed. Licensees wanted a band in which they could operate their plants and have some problems without increased regulatory response beyond the baseline inspections to provide information for indications of plant performance. To make this work, a series of thresholds were set up to serve as trigger points for increased regulatory response. The greater the degradation, the more significant the thresholds trips, and the more significant the threshold trips, the greater the regulatory response. It was emphasized that the thresholds are not predictive. They were established to suggest issues early enough to enable timely action to be taken to avoid plants going into the unacceptable region of the Action Matrix.

Dr. Bonaca indicated that it is difficult to believe that timely action can be taken if you do not have some leading indication to work from. The staff response was that they characterize timeliness as setting the thresholds low enough to trigger responses as performance problems begin to occur.

The intent in developing the thresholds was to have Green-White thresholds reflect data from experience such that only a few outliers would be in the White; White-Yellow was established such that a change of the PI would result in a delta CDF greater than 1E-05; likewise, Yellow-Red was established such that a change in the PI would result in a delta CDF greater than 1E-04. These thresholds resulted from the lower thresholds of a set of PRA models.

Examples of the unavailability thresholds were used to explain how the threshold were established. A question was asked about the generic aspects of the thresholds. The presenter stated that the licensees have a problem with the generic aspects of the thresholds because the unique features of the plant designs allow more unavailability to accrue for particular components before they get to the risk threshold than is allowed with the generic thresholds.

PI reporting by the licensees is based on unavailable hours shown in the logs from the latest quarter and from equipment histories, etc., for each piece of equipment. The data is submitted to NRC by Email and an NRC system automatically calculates the PI value using the PI definition below from NEI 99-02.

$$\text{Train unavailability} = \frac{\text{unavailable hours (planned + unplanned + fault exposure)}}{\text{Hours train required}}$$

$$\text{System unavailability} = \frac{\sum \text{Train unavailability}}{\text{Number of trains}}$$

A pilot program has just been completed regarding the scram PI because of the concerns about the unintended consequences and influences on operators in counting manual scrams. The data is in, but the results have not yet been compiled. A pilot program is also being developed for the unplanned power change PI.

Cross-cutting Issues

The final topics of discussion was cross cutting issues. These issues were defined as human performance, safety conscious work environment, and problem identification and resolution (PI&R). The assumption was that these issues would show up in the performance indicators or in the baseline inspections in a sufficient time to allow for action before safety issues arose, and therefore, did not need to be addressed separately. Every two years, a significant portion of the inspection program is directed at problem identification and resolution.

A question was raised about any given plant having set the thresholds low enough to formally identify problems. The response was that each licensee program is somewhat unique. And that if during inspections problems are identified that appear significant and the licensee did not include them in the corrective action program, then this raises the question of whether the thresholds too high or is the licensee is looking in the right areas.

Another question was raised regarding the thresholds and the requirement of the regulations. Appendix B is the applicable regulation with respect to violations. Appendix B, however, does not apply to emergency preparedness.

Initial implementation shows strong ties between plants with weak corrective action programs and plants that move out of the licensee response band to either a degraded cornerstone or a regulatory response column.

Additional actions are planned to evaluate the adequacy of the ROP as it relates to cross cutting issues.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

June 5, 2001

MEMORANDUM TO: John D. Sieber, Chairman
Plant Operations Subcommittee

FROM: Maggalean W. Weston, Senior Staff Engineer
ACRS

SUBJECT: WORKING COPY OF THE MINUTES OF THE PLANT OPERATIONS SUBCOMMITTEE HELD ON May 9, 2001, ROCKVILLE, MARYLAND

A working copy of the minutes for the subject meeting is attached for your review. Please review and comment at your earliest convenience. Copies are being sent to each ACRS Member who attended the meeting for information and/or review.

Attachment:
As Stated

cc via Email: ACRS Members
J. Larkins
J. Lyons
H. Larson
ACRS Staff and Fellows

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 10, 2001

MEMORANDUM TO: James E. Lyons, Associate Director
for Technical Support
ACRS/ACNW

FROM: Maggalean W. Weston, Senior Staff Engineer

SUBJECT: FEDERAL REGISTER NOTICE REGARDING THE MEETING OF THE ACRS SUBCOMMITTEE ON PLANT OPERATIONS, MAY 9, 2001, ROCKVILLE, MARYLAND

Attached is a Federal Register notice regarding the subject meeting. Please have this notice transmitted for publication as soon as possible.

Attachment:
FR Notice

cc with Attachment:
J. Sieber, ACRS
J. Larkins, ACRS
J. Szabo, OGC O-15D21
A. Bates, SECY O-16C1
I. Schoenfeld, OEDO O-16E15
M. Landau, OPA O-2A13
B. Boger, NRR O-6E3
W. Dean, NRR O-7A15
M. Johnson, NRR O-7A15
Public Document Room, O-1F15

[7590-01-P]

NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

MEETING OF THE ACRS SUBCOMMITTEE ON PLANT OPERATIONS

Notice of Meeting

The ACRS Subcommittee on Plant Operations will hold a meeting on May 9, 2001, in Room T-2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Wednesday, May 9, 2001 - 8:30 a.m. until the conclusion of business

The Subcommittee will discuss the Reactor Oversight Process excluding the Action Matrix. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman and written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefore, can be obtained by contacting the cognizant ACRS staff engineer, Ms. Maggalean W. Weston (telephone 301/415-3151) between 8:00 a.m. and 5:30 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Date 4/10/01

J. E. Lyons

James E. Lyons, Associate Director
for Technical Support

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
PLANT OPERATIONS SUBCOMMITTEE
May 9, 2001
ROCKVILLE, MARYLAND

-AGENDA-

	SUBJECT	PRESENTER	TIME
I.	Introductory Remarks Subcommittee Chair	J. Sieber	8:30-8:35 a.m.
II.	NRC Staff Presentation		8:35-10:00 a.m.
	Introduction	- Mike Johnson, NRR	
	Significance Determination	- Doug Coe, NRR - J.S. Hyslop, NRR - Mark Salley, NRR	
	Performance Indicators	- Don Hickman, NRR	
	Cross-cutting Issues	- Jeff Jacobson, NRR	
		************BREAK************	10:00-10:20 a.m.
III.	NRC Staff Presentation (Continued)		10:20-12:00 noon
		************LUNCH************	12:00-12:45 p.m.
IV.	NRC Staff Presentation (Continued)		12:45-2:00 p.m.
V.	General Discussion and Adjournment		2:00-2:30 p.m.

NOTE: Number of copies of the presentation materials to be provided to the ACRS is 35

ACRS CONTACT: Ms. Maggalean W. Weston, mww@nrc.gov or (301) 415-3151.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON THE PLANT OPERATIONS

MAY 9, 2001
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
John Hannon	SPLB/DSSA
Michael Johnson	~~NR~~ IIPB/DIPM
Don Hickman	IIPB/DIPM
David Allsopp	IIPB/DIPM
Peter Wilson	SPSB/DSSA
See-Meng Wong	NRR/ADIP
J.S. Hyslop	NRR/SPSB
Eugene Cobey	NRR/DIPM
John Thompson	NRR/DIPM
Mark Satorius	NRR/DIPM
Mark Henry Salley	NRR/DSSA/SPLB
Steven Stein	NRR/DIPM
Tom Boyce	NRR/DIPM/IIPB
A. El-Bassioni	NRR DSSA/SPSB
J Jolicoeur	NRR/DIPM
Phil Qualls	NRR/DSSA/SPLB
Leon Whitney	NRR/DIPM/IIPB
Peter Koltay	NRC DIPM
Garth Parry	NRR/DSSA
Alan Madison	NRR/IIPB

ACTION: Notice of meeting.

SUMMARY: NRC will host a public meeting in Rockville, Maryland. The meeting will provide an opportunity for discussion on the revised draft Chapter 3 entitled, "Integrated Safety Analysis" of NUREG–1520 for 10 CFR part 70, Standard Review Plan (SRP) for the Review of a License Application for a Fuel Cycle Facility. The March 30, 2001, draft Chapter 3 can be found in both a "clean" and marked-up version in the NRC Public Electronic Reading Room under "Recently Released Documents, April 3, 2001". It can also be found on the Internet at the following website:

http://techconf.llnl.gov/cgi-bin/library?source=*&library=Part_70_lib

The web site can also be reached by the following method:

1. Go to the main NRC web site at: http://www.nrc.gov.

2. Scroll down to the bottom of that page and click on the word "Rulemaking."

3. Scroll down on the Rulemaking page until the words "Technical Conference" appear. Click on those words.

4. On the page titled "Welcome to the NRC Technical Conference Forum," click on the link "Conference" or "Technical Conferences".

5. Scroll down to the topic "Draft Standard Review Plan and Guidance on Amendment to 10 CFR Part 70."

6. Select "Document Library."

Purpose: This meeting will provide an opportunity to discuss comments on the staff's revised draft Chapter 3 and its appendix.

DATES: The meeting is scheduled for Tuesday, May 8, 2001, from 1:00 p.m. to 4:00 p.m. The meeting is open to the public.

ADDRESSES: Two White Flint North, 11545 Rockville Pike, Room T–10A1, Rockville, Maryland. Visitor parking around the NRC building is limited; however, the meeting site is located adjacent to the White Flint Station on the Metro Red Line.

FOR FURTHER INFORMATION CONTACT: Yawar H. Faraz, Senior Project Manager, Fuel Cycle Licensing Branch, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone: (301) 415–8113, e-mail *yhf@nrc.gov.*

Dated at Rockville, Maryland this 9th day of April, 2001.

For the Nuclear Regulatory Commission.

Lidia Roché,

Acting Chief, Fuel Cycle Licensing Branch, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 01–9319 Filed 4–13–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Subcommittee Meeting on Planning and Procedures; Notice of Meeting

The ACRS Subcommittee on Planning and Procedures will hold a meeting on May 9, 2001, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b(c)(2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows:

Wednesday, May 9, 2001—2:30 p.m. Until the Conclusion of Business

The Subcommittee will discuss proposed ACRS activities and related matters. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff person named below five days prior to the meeting, if possible, so that appropriate arrangements can be made..

Further information regarding topics to be discussed, the scheduling of sessions open to the public, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements, and the time allotted therefor can be obtained by contacting the cognizant ACRS staff person, Dr. John T. Larkins (telephone: 301/415–7360) between 7:30 a.m. and 4:15 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any changes in schedule, etc., that may have occurred.

Dated: April 9, 2001.

James E. Lyons,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 01–9321 Filed 4–13–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting of the ACRS Subcommittee on Plant Operations; Notice of Meeting

The ACRS Subcommittee on Plant Operations will hold a meeting on May 9, 2001, in Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Wednesday, May 9, 2001—8:30 a.m. Until the Conclusion of Business

The Subcommittee will discuss the Reactor Oversight Process excluding the Action Matrix. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman and written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefore, can be obtained by contacting the cognizant ACRS staff engineer, Ms. Maggalean W. Weston (telephone 301/415–3151) between 8 a.m. and 5:30 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: April 10, 2001.

James E. Lyons,

Associate Director for Technical Support.

[FR Doc. 01–9322 Filed 8–13–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission has issued a new guide in its Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

Regulatory Guide 1.189, "Fire Protection for Operating Nuclear Power Plants," has been developed to provide a comprehensive fire protection guidance document and to identify the scope and depth of fire protection that the NRC staff has determined to be acceptable for operating nuclear plants.

Comments and suggestions in connection with items for inclusion in guides currently being developed or improvements in all published guides are encouraged at any time. Written comments may be submitted to the Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Regulatory guides are available for inspection or downloading at the NRC's web site at <*WWW.NRC.GOV*> under Regulatory Guides and in NRC's Electronic Reading Room (ADAMS System) at the same site; Regulatory Guide 1.189 has Accession Number ML010920084. Single copies of regulatory guides may be obtained free of charge by writing the Reproduction and Distribution Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, or by fax to (301)415–2289, or by email to <*DISTRIBUTION@NRC.GOV*>. Issued guides may also be purchased from the National Technical Information Service on a standing order basis. Details on this service may be obtained by writing NTIS, 5285 Port Royal Road, Springfield, VA 22161. Regulatory guides are not copyrighted, and NRC approval is not required to reproduce them.

(5 U.S.C. 552(a))

Dated at Rockville, Maryland, this 6th day of April 2001.

For the Nuclear Regulatory Commission.

Roy P. Zimmerman,

Deputy Director, Office of Nuclear Regulatory Research.

[FR Doc. 01–9317 Filed 4–13–01; 8:45 am]

BILLING CODE 7590–01–P

RAILROAD RETIREMENT BOARD

Proposed Collection; Comment Request

SUMMARY: In accordance with the requirement of Section 3506 (c)(2)(A) of the Paperwork Reduction Act of 1995 which provides opportunity for public comment on new or revised data collections, the Railroad Retirement Board (RRB) will publish periodic summaries of proposed data collections.

Comments are invited on: (a) Whether the proposed information collection is necessary for the proper performance of the functions of the agency, including whether the information has practical utility; (b) the accuracy of the RRB's estimate of the burden of the collection of the information; (c) ways to enhance the quality, utility, and clarity of the information to be collected; and (d) ways to minimize the burden related to the collection of information on respondents, including the use of automated collection techniques or other forms of information technology.

Title and Purpose of Information Collection

Certification Regarding Rights to Unemployment Benefits; OMB 3220–0079

Under Section 4 of the Railroad Unemployment Insurance Act (RUIA), an employee who leaves work voluntarily is disqualified for unemployment benefits unless the employee left work for good cause and is not qualified for unemployment benefits under any other law. RRB Form UI–45, Claimant's Statement—Voluntary Leaving of Work, is used by the RRB to obtain additional information needed to investigate a claim for unemployment benefits when the claimant indicates on RRB Form UI–1, Application for Unemployment Benefits and Employment Service (OMB 3220–0022) that he has voluntarily left work. Completion of Form UI–45 is required to obtain or retain benefits. One response is received from each respondent.

RRB Form UI–45 is being revised to include language that asks a claimant, if they have been denied state unemployment benefits or unemployment benefits under any law other than the RUIA, to attach a copy of any letter received that denied them the benefits they applied for. Minor non-burden impacting editorial changes are also being proposed. The completion time for the UI–45 is estimated at 15 minutes per response. The RRB estimates that approximately 2,900 responses are received annually.

Additional Information or Comments: To request more information or to obtain a copy of the information collection justification, forms, and/or supporting material, please call the RRB Clearance Officer, at (312) 751–3363. Comments regarding the information collection should be addressed to Ronald J. Hodapp, Railroad Retirement Board, 844 North Rush Street, Chicago, Illinois 60611–2092. Written comments should be received within 60 days of this notice.

Chuck Mierzwa,

Clearance Officer.

[FR Doc. 01–9290 Filed 4–13–01; 8:45 am]

BILLING CODE 7905–01–M

RAILROAD RETIREMENT BOARD

Proposed Collection; Comment Request

SUMMARY: In accordance with the requirement of Section 3506(c)(2)(A) of the Paperwork Reduction Act of 1995 which provides opportunity for public comment on new or revised data collections, the Railroad Retirement Board (RRB) will publish periodic summaries of proposed data collections.

Comments are invited on: (a) Whether the proposed information collection is necessary for the proper performance of the functions of the agency, including whether the information has practical utility; (b) the accuracy of the RRB's estimate of the burden of the collection of the information; (c) ways to enhance the quality, utility, and clarity of the

STATUS OF THE REACTOR OVERSIGHT PROCESS

Inspection Program Branch
May 9, 2001

UNITED STATES NUCLEAR REGULATORY COMMISSION

TOPICS FOR DISCUSSION

- **Initial Implementation Status**
- **Significance Determination Process**
- **Performance Indicators**
- **Selected Issues**

OVERALL RESULTS

- Substantially Exercised Process
- Made Several Significant Changes
- Maintained Process Stable
- Successful Demonstration of Framework Objectives
- Data on Process Results and Resources

Example SDPs

Two No-color Findings

One Green Finding

One Non-Green Finding

Example No-color Inspection Finding

No-color Findings: Findings which do not affect a cornerstone or which have extenuating circumstances

Inspection Source: Inspection of issues identified in licensee LERs using Inspection Procedure 71153 (Inspection Report 05000286/2000-04)

Finding: Missed Control Room Oxygen Detector Surveillance Tests

Example No-color Inspection Finding (continued)

Documentation: Per IMC 0610* Appendix B, a finding that does not affect a cornerstone and cannot be processed by an SDP is documented as a No-color finding

Disposition of Finding: Confirmed entry into licensee corrective action program and treated as a non-cited violation (NCV) consistent with the NRC Enforcement Policy

Example No-color Inspection Finding

No-color Findings: Findings which do not affect a cornerstone or which have extenuating circumstances

Inspection Source: Inspection of licensee's Problem Identification and Resolution (PI&R) program using Inspection Procedure 71152 (Inspection Report 05000277/2000-13)

Finding: One Operator License application was submitted to the NRC incorrectly stating that certain training had been completed

Example No-color Inspection Finding (continued)

Documentation: Per IMC 0610* Appendix B, a finding that potentially impacts the NRC's ability to perform its regulatory function (e.g., failure to provide complete and accurate information) is documented as a No-color finding

Disposition of Finding: Confirmed entery into licensee corrective action program and treated as a non-cited violation consistent with the NRC Enforcement Policy

Example Green Inspection Finding

Inspection Source: Inspection of Surveillence Testing using Inspection Procedure 71111.22 (Inspection Report 05000260/2000-06)

Finding: Inadequate evaluation of RHR system flow rate test results. No evaluation was performed following temporary modifications which reduced RHR system flow rate to ensure that Technical Specification flow rate requirements were satisfied. Subsequent evaluations showed that system flow met all surveillence test requirements.

Example Green Inspection Finding (continued)

Documentation: Meets IMC 0610* documentation threshold based on affecting the mitigation systems cornerstone

Disposition of Finding: Confirmed entry into licensee corrective action program. Phase 1 SDP: Green finding based on system function affected, but operability and function maintained.

Example White Inspection Finding

Finding: Oil leak rate on one Emergency Feedwater (EFW) Pump bearing forced operators to make daily oil additions to maintain a visible level in the oiler sightglass. Upon questioning by resident inspectors, licensee determined that all four bolts on the outboard bearing housing inner cover had been loose for 39 days and that the pump would have overheated its bearing within a few hours of operation.

Example White Inspection Finding (continued)

Documentation: Meets IMC 0610* documentation threshold based on affecting the mitigation systems cornerstone

Phase 1 SDP

Continue to Phase 2 based on actual loss of safety function of a single train for greater than its Tech Spec Allowed Outage Time.

Example White Inspection Finding (continued)

Phase 2 SDP

Dominant accident sequence

Transient with Loss of PCS - Loss of EFW - Loss of Feed/Bleed

Initiating Event Likelihood = "A"
(SDP Table 1 using IE frequency > 0.1/yr and exposure time > 30 days)

Mitigating System Credit = 2 (remaining MDEFW pump)
1 (for available TDEFW pump)
2 (for feed/bleed availability)
5 (total credit - no operator recovery)

Significance Result = White (1E-6/yr to 1E-5/yr)

Example White Inspection Finding (continued)

Phase 3 SDP

Licensee PRA result for unrecoverable loss of MDEFW pump for 39 days is a change in CDF of 2.7E-6/yr

Neither result includes external event contributions (considered not to increase risk contribution beyond White, based on available external event risk insights)

Enforcement Action

Notice of Violation of 10CFR Part 50, Appendix B, Criterion XVI, “Corrective Action” would be issued as is appropriate for an inspection finding of White or greater significance.   Formal licensee response is required.

SDP PHASE 1 SCREENING WORKSHEET FOR IE, MS, and B CORNERSTONES

Reference/Title: NRC Inspection Report [redacted]
Motor Driven Emergency Feedwater Pump Inoperaable Due to Oil Leak

Factual Description of Identified Condition: On February 12, 2001, the resident inspectors found the oiler on the 'A' motor driven emergency feedwater pump (EF-P-2A) outboard bearing empty. An oil leak developed on the pump outboard bearing during performance of a quarterly surveillance run of the pump on February 1, 2001. The leak rate was of sufficient magnitude to require auxiliary operators to make daily oil additions to the pump bearing to maintain a visible level in the oiler sightglass. On February 14, 2001, engineers identified all four bolts on the outboard bearing housing inner cover were loose. Two of the four bolts were tightened one full turn each, and the other two one-half turn each. Tightening the bolts stopped the oil leak and corrected an unexplained step increase in pump vibrations observed during the February 1, 2001 surveillance test.

System(s)affected by identified condition: Emergency Feedwater System

Train(s) affected by identified condition: 'A' motor driven emergency feedwater pump train

Licensing Basis Function of System(s) or Train(s) (as applicable): The emergency feedwater system consists of three independent pumps (two motor driven and one turbine driven) and associated flowpaths to each of two steam generators. The system is designed to automatically actuate and remove secondary heat when the main feedwater system fails to function. Two of three pumps are required to meet the most limiting design basis flow requirement. The technical specification limiting condition for operation requires three pumps to be operable and permits an allowed outage time of 72 hours for one inoperable pump. An inservice test of each pump is required every 92 days.

Other Safety Function of System(s) or Train(s) (as applicable): None

Maintenance Rule category (check one): risk-significant

Time that identified condition existed or is assumed to have existed: Prior to the February 1, 2001 surveillance test, EF-P-2A was last run on January 6, 2001, for testing of the emergency feedwater system automatic start circuit. Any time after January 6, 2001, that EF-P-2A was called on to operate, would have resulted in the same condition that occurred following the February 1, 2001 surveillance test. Namely, the bearing housing cover bolts would have loosened and an excessive oil leak would have developed. Based on the observed rate of oil loss following the February 1, 2001 surveillance test, no credit is given for operator action to maintain the pump operable. The pump was inoperable from January 6, 2001, until the pump was repaired on February 14, 2001, a period of 39 days.

Functions and Cornerstones affected as a result of this identified condition (check ✓)

INITIATING EVENT CORNERSTONE

_____ Transient initiator contributor (e.g., reactor/turbine trip, loss offsite power)

_____ Primary or Secondary system LOCA initiator contributor (e.g., RCS or main steam/feedwater pipe degradations and leaks)

MITIGATION SYSTEMS CORNERSTONE

_____ Core Decay Heat Removal Degraded

_____ Initial Injection Heat Removal Degraded

_____ Primary (e.g., Safety Inj)

____ Low Pressure

____ High Pressure

__X___ Secondary - PWR only (e.g., AFW)

_____ Long Term Heat Removal Degraded (e.g., ECCS sump recirculation, suppression pool cooling)

_____ Reactivity Control Degraded

_____ Fire/Flood/Seismic/WeatherProtection Degraded

BARRIERS CORNERSTONE

_____ RCS LOCA Mitigation Boundary Degraded (e.g., PORV block valve, PTS issue)

_____ Containment Barrier Degraded

_____ Reactor Containment Degraded

_____ Actual Breach or Bypass

_____ Heat Removal, Hydrogen or Pressure Control Degraded

_____ Control Room, Aux Bldg, or Spent Fuel Bldg Barrier Degraded

_____ Fuel Cladding Barrier Degraded

SDP PHASE 1 SCREENING WORKSHEET FOR IE, MS, and B CORNERSTONES

Seismic, Fire, Flooding, and Severe Weather Screening Criteria

1. Does the finding involve the loss or degradation of equipment or function **specifically** designed to mitigate a seismic, flooding, or severe weather initiating event (e.g., seismic snubbers, flooding barriers, tornado doors)? (Equipment and functions for the mitigation or suppression of fire initiating events, such as thermal wrap or sprinkler systems, should be evaluated using IMC 0609 Appendix F and are not evaluated here)

   ☐ **If YES** → continue to question 2

   ☐ If NO → skip to question 3

2. If the safety function is assumed to be completely failed or unavailable, are ANY of the following three statements TRUE? The loss of the affected equipment or function by itself, during the external initiating event it was intended to mitigate

   a) would cause a plant trip or any of the Initiating Events used by Phase 2 for the plant in question;

   b) would degrade **more than** a single Train of a multi-train safety system or function;

   c) would degrade the function of any one Train of a support system for a safety system or function.

   ☐ **If YES** → the finding is potentially risk significant due to external initiating event core damage sequences - return to page 2 of this Worksheet

   ☐ If NO, screen as Green

3. Does the finding involve the loss of any safety function, identified by the licensee through a PRA, IPEEE, or similar analysis, that contributes to external event initiated core damage accident sequences (i.e., initiated by a seismic, fire, flooding, or severe weather event)?

   ☐ **If YES** → the finding is potentially risk significant due to external initiating event core damage sequences - return to page 2 of this Worksheet

   ☐ If NO, screen as Green

Result of Phase 1 screening process: ☐ screen as Green **X go to Phase 2** ☐ input to Phase 3

Important Assumptions (as applicable):

SDP PHASE 1 SCREENING WORKSHEET FOR IE, MS, and B CORNERSTONES Check the appropriate boxes ✔			
If the finding is assumed to affect: 1. fire barrier or suppression features, use IMC 0609 Appendix F 2. the safety of a shutdown reactor, use IMC 0609 Appendix G 3. the safety of an operating reactor, identify the affected areas: □Initiating Event **X** Mitigation Systems □RCS Barrier □Fuel Barrier □Containment Barriers 4. **None** of the above areas affected → screen as Green 5. **Two or more** of the above areas affected → **Go to Phase 2** 6. If **only one** of the above areas is affected, continue **only** in the **appropriate** column below.			
Initiating Event 1. Does the finding contribute to the likelihood of a Primary or Secondary system LOCA initiator? □**If YES → Go to Phase 2** □If NO, continue 2. Does the finding contribute to both the likelihood of a reactor trip AND the likelihood that mitigation equipment or functions will not be available? □**If YES → Go to Phase 2** □If NO, continue 3. Does the finding increase the likelihood of a fire or internal/external flood? □**If YES →** Use the IPEEE or other existing plant-specific analyses to identify core damage scenarios of concern and factors that increase the frequency. Provide this input for **Phase 3** analysis. □If NO, screen as Green	Mitigation Systems 1. Is the finding a design or qualification deficiency that does NOT affect operability per GL 91-18 (rev 1)? □**If YES →** screen as Green **X** If NO, continue 2. Does the finding represent an actual loss of safety function of a System? □**If YES → Go to Phase 2** **X** If NO, continue 3. Does the finding represent an actual loss of safety function of a single Train, for > its Tech Spec Allowed Outage Time? **X If YES → Go To Phase 2** □If NO, continue 4. Does the finding represent an actual loss of safety function of one or more non-Tech Spec Trains of equipment designated as risk-significant per 10CFR50.65, for >24 hrs? □**If YES → Go To Phase 2** □If NO, continue 5. Does the finding screen as potentially risk significant due to a seismic, fire, flooding, or severe weather initiating event, using the criteria on page 3 of this Worksheet? □**If YES →** Use the IPEEE or other existing plant-specific analyses to identify core damage scenarios of concern and provide this input for **Phase 3** analysis. □If NO, screen as Green	RCS Barrier or Fuel Barrier 1. RCS Barrier **Go to Phase 2** 2. Fuel Barrier screen as Green	Containment Barriers 1. Does the finding represent a degradation of the radiological barrier function provided for the control room, auxiliary building, or spent fuel? □**If YES →** screen as Green □If NO, continue 2. Does the finding represent a degradation of the barrier function of the control room against smoke or a toxic atmosphere? □**If YES → Go to Phase 3** □If NO, continue 3. Does the finding represent an actual open pathway in the physical integrity of reactor containment or an actual reduction of the atmospheric pressure control function of the reactor containment? □**If YES → Go to Phase 2 in Appendix H of IMC 0609** □If NO, screen as Green

Table 1 Categories of Initiating Events for Nuclear Generating Station, Unit 1

Row	Approximate Frequency	Example Event Type	Estimated Likelihood Rating		
I	> 1 per 1-10 yr	Reactor Trip (TRANS), Loss of Power Conversion System (TPCS)	A	B	C
II	1 per 10-10^2 yr	Loss of offsite power (LOOP)	B	C	D
III	1 per 10^2 - 10^3 yr	SGTR, Stuck open PORV/SRV (SORV), Small LOCA including RCP seal failures (SLOCA), Main Steam Line Break (MSLB), Loss of Instrument Air (LIA), Loss of a L4kVBUS (L4kVBUS), Loss of Nuclear Service River Water (LNSRW)	C	D	E
IV	1 per 10^3 - 10^4 yr	Medium LOCA (MLOCA), LOOP with loss of one division of emergency AC (LOOP1EDG), Loss of River Water (LRW)	D	E	F
V	1 per 10^4 - 10^5 yr	Large LOCA (LLOCA)	E	F	G
VI	less than 1 per 10^5 yr	ATWS (mechanical only), ISLOCA	F	G	H
			> 30 days	3-30 days	< 3 days
			Exposure Time for Degraded Condition		

Note:

1. The SDP worksheets for ATWS core damage sequences assume that the ATWS is not recoverable by manual actuation of the reactor trip function or by ARI for BWRs. Thus, the ATWS frequency to be used by these worksheets must represent the ATWS condition that can only be mitigated by the systems shown in the worksheet (e.g., boration). Any inspection finding that represents a loss of manual reactor trip capability for a postulated ATWS scenario should be evaluated by a risk analyst for consideration of the probability of a successful manual trip.

Table 2 Initiators and System Dependency for Nuclear Generating Station, Unit 1 [1]

Affected Systems	Major Components	Support Systems	Initiating Event
AC Power System	AC Power Distribution & AC Instrument Power	DC	All
EFW	2 MDPs	AC , DC	All except MLOCA, LLOCA and LIA
	1 TDP	Vital AC, DC	
Fire Service Water System[2]	3 pumps, 2 taking suction from river, 1 from circulating water flume	AC(FS-P2 only), DC(FS-P2 only)	LOOP, LRW
Nuclear Service Closed Cooling Water (NSCCW)	3 50% pumps and 4 1/3 capacity heat exchangers	AC, DC, NSRW	All except LLOCA and LNSRW
Nuclear Service River Water System (NSRW)	3 50% pumps	AC, DC	LNSRW
Condensate / MFW	Three Condensate pumps	AC, DC	TRANS, L4kVBUS, SGTR, LNSRW
	Two TDMFW Pumps	Vital instrument bus, DC, IA, SSCCW	
Decay Heat Closed Cooling Water System (DHCCW)	2 pumps	AC, DC, DHRW backed up by fire service water system	All
Decay Heat River Water System (DHRW)	2 pumps	AC, DC	All
HPI / Makeup and Purification	3 HPI pumps (300 gpm at 2400 psi) in 2 injection trains	AC, DC, NSCCW (pump B cooling), DHCCW (Pumps A and C only, backed up by NSCCW)	All except LLOCA and LRW
DC Power System	2 Buses, 2 battery chargers, and two batteries	6 hours battery depletion time	All

Table 2 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event
Emergency AC (EDG)	2 EDGs, 1 SBO DG	DC, Fire service water (for SBO DG), Fuel transfer pumps	LOOP, LOOP1EDG
Instrument Air (IA)	IA: 3 air compressors	AC, DC, SSCCW backup by fire service water	LIA
Intermediate Closed Cooling Water System (ICCW)	2 Pumps	AC, DC, NSRW (heat exchangers), IA	All except LNSRW, LIA, and LRW
Main Steam	Per SG: 1 ADV, 2 steam lines, 10 safety valves, 2 MSIVs and 3 turbine bypass valves (capacity 21.4%)	DC, IA, AC (for MSIVs)	All except MLOCA and LLOCA
Pressurizer Pressure Relief and Pressurizer Venting for SGTR depressurization	2 Pressurizer Safety valves, 1 PORV with associated block valve	AC (block valve), vital AC, DC (PORV)	All except MLOCA, LLOCA and LRW
	1 Pressurizer vent line with 1 MOV and 1 SOV in series	Class 1E AC train B, Class B DC train B	
RCP	Westinghouse Seals	3 HPI pumps for seal injection, ICCW for thermal barrier cooling, and IA for injection valve MU-V20, and ICCW valves IC-V3 and V4, NSCW (RCP motors)	SLOCA
LPI / DHR	2 RHR/LPSI pumps and heat exchangers	AC, DC, DHCCW, DHRW, ESFAS	All except ATWS and LNSRW
Secondary Service Closed Cooling Water System (SSCCW)	3 50% pumps and 4 1/3 capacity heat exchangers	AC, DC, SSRW	LIA, TRANS, L4kVBUS, SGTR, LNSRW
Secondary Service River Water System (SSRW)	3 pumps	AC, DC	Same as SSCCW

Table 2 (Continued)

Notes:

1. Plant internal event CDF = 4.1 E-5/yr, including contribution 3.E-6 from internal floods. (See page 10-6).

2. Fire service water system is included in maintenance rule scope.

Initiating Event Likelihood	Remaining Mitigation Capability Rating (with Examples)						
	6	5	4	3	2	1	0
	3 diverse trains OR 2 multi-train systems OR 1 train + 1 multi-train system + recovery of failed train	1 train + 1 multi-train system OR 2 diverse trains + recovery of failed train	2 diverse trains OR 1 multi-train system + recovery of failed train	1 train + recovery of failed train OR 1 multi-train system OR Operator action + recovery of failed train	1 train OR Operator action OR Operator action under high stress + recovery of failed train	Recovery of failed train OR Operator action under high stress	none
A	Green	White	Yellow	Red	Red	Red	Red
B	Green	Green	White	Yellow	Red	Red	Red
C	Green	Green	Green	White	Yellow	Red	Red
D	Green	Green	Green	Green	White	Yellow	Red
E	Green	Green	Green	Green	Green	White	Yellow
F	Green	Green	Green	Green	Green	Green	White
G	Green	Green	Green	Green	Green	Green	Green
H	Green	Green	Green	Green	Green	Green	Green

Table 4 - Risk Significance Estimation Matrix

Table 3.2 SDP Worksheet for Nuclear Generating Station Unit 1 — Transients with Loss of PCS (TPCS)

Estimated Frequency (Table 1 Row) I **Exposure Time** >30days **Table 1 Result (circle): A**

Safety Functions Needed:	Full Creditable Mitigation Capability for Each Safety Function:
Secondary Heat Removal (EFW)	1/2 MDEFW trains (1 multi-train system) or 1/1 TDEFW train (1 ASD train) with steam relief through 1/2 ADVs or 1/20 safety valves
Primary Bleed (FB)	1/2 SRVs open (1 multi-train system) or (1/1 PORV open or 1/1 Pressurizer vent line open) (operator action = 2) (1)
Early Inventory, HPI Injection (EIHP)	1/2 HPI trains (3 pumps) injecting from BWST (1 multi-train system)
High Pressure Recirculation (HPR)	1/2 HPI trains (3 pumps) taking suction from 1/2 LPI trains (operator action = 3) (2)

Circle Affected Functions	Recovery of Failed Train	Remaining Mitigation Capability Rating for Each Affected Sequence	Sequence Color
1. TPCS - EFW - HPR (3)		2(MDEFW)+1(TDEFW)+3(HPR)=6	GR/NTW
2 TPCS - EFW - EIHP (4)		2(MDEFW)+1(TDEFW)+3(EIHP)=6	GR/NTW
3 TPCS - EFW - FB (5)		2(MDEFW)+1(TDEFW)+2(FB)=5	WHITE

Notes:
1. The HEP used in the IPE is 3.44E-2 for the failure of the operator to manually establish HPI cooling (feed and bleed). (Event HBW1, Table 6.1-2.)
2. Operator action is required to establish recirculation. The HEP assessed in the IPE for switchover to recirculation is 1.27E-4 (event HSR2, Table 6.1-2) which is lower than the generic credit of 3 based on the HEPs of XXX plants. In this worksheet, the generic credit is used.

Table 3.7 SDP Worksheet for Nuclear Generating Station Unit 1 — Loss of Offsite Power (LOOP) [1]

Estimated Frequency (Table 1 Row) II **Exposure Time** >30 days **Table 1 Result (circle): B**

Safety Functions Needed:	**Full Creditable Mitigation Capability for Each Safety Function:**
Emergency AC Power (EAC)	1/2 EDGs (1 multi-train system) or SBO DG (operator action = 1) [2]
Turbine-driven EFW Pump (TDEFW)	1/1 TDEFW train (1 ASD train) with steam relief through 1/2 ADVs or 1/20 safety valves
Recovery of AC Power in < 1 hrs (REC1)	Recovery of an AC source (operator action = 1) [3]
Recovery of AC Power in < 4 hrs (REC4)	Recovery of an AC source (operator action = 2) [4]
Secondary Heat Removal (MDEFW)	1/2 MDEFW train after AC recovered-Excluding TDEFW credited earlier (1 multi-train system) with steam relief through 1/2 ADVs or 1/20 safety valves
Secondary Heat Removal (EFW)	1/2 MDEFW trains (1 multi-train system) or 1/1 TDEFW train (1 ASD train)
High Pressure Injection (HPI)	1/2 HPI trains (3 pumps) injecting from BWST (1 multi-train system)
Primary Bleed (FB)	1/2 SRVs open (1 multi-train system) or (1/1 PORV open or 1/1 Pressurizer vent line open) (operator action = 2) [5]
High Pressure Recirculation (HPR)	1/2 HPI trains (3 pumps) taking suction from 1/2 LPI trains (operator action = 3) [6]

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LOOP - EFW - HPR (1) [6]		2(MDEFW)+1(TDEFW)+3(HPR)=6	GR
2 LOOP - EFW - HPI (1)		2(MDEFW)+1(TDEFW)+3(HPI)=6	GR
3 LOOP - EFW - FB (1)		2(MDEFW)+1(TDEFW)+2(FB)=5	GR/NTW
4 LOOP - EAC - REC1 - HPR (4) (AC recovered, Seal LOCA)			
5 LOOP - EAC - REC1 - HPI (5) (AC recovered, Seal LOCA)			

6 LOOP - EAC - REC4 (6) (SBO, Seal LOCA)			
7 LOOP - EAC - TDEFW - MDEFW - HPR (9) (AC recovered)		3(EAC)+1(SBO)+2(MDEFW)+1(TDEFW)+3(HPR)=10	GR
8 LOOP - EAC - TDEFW - MDEFW - FB (10) (AC recovered)		3(EAC)+1(SBO)+2(MDEFW)+1(TDEFW)+2(FB)=9	GR
9 LOOP - EAC - TDEFW - MDEFW - HPI (11) (AC recovered)		3(EAC)+1(SBO)+2(MDEFW)+1(TDEFW)+3(HPI)=10	GR
10 LOOP - EAC - TDEFW - REC1 (12)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

1. The frequency of LOOP is 1.92E-2 / year. The IPE does not explain how seal LOCA is considered in a SBO. In this worksheet, it is assumed that in a SBO a RCP seal LOCA occurs at 1 hour and core damage takes place at 4 hours.

2. Based on generic consideration, a credit of 1 is given to this operator action. The HEP for operator failure to start SBO DG and connect it to an emergency bus is 2.34E-3. (Event HEO1A, Table 6.1-2.)

3. The probability of operator failure to recover AC power, given failure of TDEFW, is 8.46E-2. (Event HRE3, Table 6.1-2.)

4. The probability of operator failure to recover AC power, given TDEFW initially available, is 8.31E-3. (Event HRE1, Table 6.1-2.)

5. The HEP used in the IPE is 3.44E-2 for the failure of the operator to manually establish HPI cooling (feed and bleed). (Event HBW1, Table 6.1-2.)

6. Operator action is required to establish recirculation. The HEP assessed in the IPE for switch over to recirculation is 1.27E-4 (event HSR2, Table 6.1-2) which is lower than the generic credit of 3 based on the HEPs of XXX plants. In this worksheet, the generic credit is used.

FIRE PROTECTION SIGNIFICANCE DETERMINATION PROCESS

by

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Given to ACRS, May 9, 2001

OVERVIEW

- General Remarks
- Example: Evaluation of specific fire protection findings
  - Clear identification of findings
  - Configuration of room in which findings exist
  - Applying fire protection SDP to estimate risk (color findings)
    - Basis for degradation levels and failure probabilities

GENERAL REMARKS

- Uses PRA techniques and data generally accepted by fire risk community
- Is an evolving process
- Communicating with stakeholders
  - Fire Protection Information Forum
  - Reactor Oversight Process Workshop
- Coordinating with Office of Research - fire research plan

EXAMPLE

- Clear identification of findings
  - Manual fixed fire suppression system (CO_2) would not maintain minimum concentration for the fire hazard
  - Electrical raceway fire barrier system protecting redundant trains did not meet one hour rating
  - Each finding existed simultaneously for greater than thirty days

EXAMPLE: phase 2

- Configuration of room in which findings exist
    - 4160V Essential Switchgear room divided into 3 sections by two partial-height marinite walls
    - Each section contains a 4160V Vital Bus, where two electrical buses needed to support one mechanical train
    - Cables from all electrical trains come together in the overhead of the center section in vicinity of 4160V Vital Switchgear cabinet
- Fire scenario: fire starts in 4160V Vital Switchgear cabinet in center section ; fire plume damages cabling from both remaining electrical trains

EXAMPLE: phase 2

- Evaluate FMF (fire mitigation frequency) = IF + AS + MS + FB + CC
  - Determine the fire ignition frequency (IF) of the 4160V Vital Switchgear cabinet in center bay - IPEEE
  - Evaluate extent of degradations identified in fixed suppression (AS) and the fire barrier (FB) - choice of moderate or high for either
  - No degradation in manual suppression, i.e. fire brigade, identified (MS); no dependency/common cause (CC) contribution

EXAMPLE: phase 2 (cont.)

- Degradations
  - Moderate degradation for auto-suppression system (AS)
    - Manual actuation
    - CO_2 system only reached 46% concentration (minimum required was 50%)
  - Moderate to high degradation for one hour fire barrier (FB)
    - Tests indicate the barrier had a rating of 10 to 15 minutes
- No degradation in fire brigade (normal operating state)
  - Professionally trained fire brigade, performed drill satisfactorily and arrived within 11 minutes of alarm notification

EXAMPLE: phase 2 (cont.)

- Evaluate FMF (fire mitigation frequency) = IF + AS + MS + FB + CC
  - Determine the fire ignition frequency (IF) of the 4160V Vital Switchgear cabinet in center bay - IPEEE
  - Include extent of degradations identified in fixed suppression (AS) and the fire barrier (FB)
  - No degradation in manual suppression, i.e. fire brigade, identified (MS); no dependency/common cause (CC) contribution

- FMF = -3 - 0.75 - 0.25 - 1 = -5 since
  - Fire ignition frequency = $\log_{10}(1E\text{-}3) = -3$
  - Moderate degradation for autosuppression (AS) = -0.75
  - Moderate to high degradation for one hour barrier (FB) = -0.25
  - Normal operating state for fire brigade (MS) = -1

EXAMPLE: phase 2 (cont.)
FAILURE PROBABILITIES FOR DID

Level of Degradation	3 Hour Barrier	1 Hour Barrier	Auto-Suppress	Fire Brigade
High	0	0	0	-0.25
Moderate	-1.25	-0.5	-0.75	-0.5
Normal Operating State (NOS)	-2 (door) -2.5 -3	-1	-1.25	-1

EXAMPLE: phase 2 (cont.)

- Utilize reactor safety worksheets
  - Fire induces a SLOCA - RCP Seal LOCA
    - Loss of CCW and charging with no high pressure injection available
  - Core damage - no credit for mitigating systems
- Resulting evaluation is white
- Higher degradation in any DID element produces a yellow; repairing fire barrier produces green/white threshold.

PHASE 2 RISK ESTIMATION WORKSHEET Small LOCA

Estimated Frequency (Table 1 Row) __-5__	Exposure Time __> 30 DAYS__	Table 1 Result (circle): A B C D E F G H	
Safety Functions Needed:	*Full* Creditable Mitigation Capability for each Safety Function:		
Early Inventory, HP Injection (EIHP)	1 / 4 Charging or SI trains (2 multi-train systems)		
Power Conversion System (PCS)	1/3 condensate pump (operator action)		
Secondary Heat Removal (AFW)	1 / 2 MDAFW trains (1 multi-train system) or 1 TDAFW train (1 ASD train)		
Primary Heat Removal, Feed/Bleed (FB)	1 / 2 PORVs open for Feed/Bleed (operator action)		
High Pressure Recirculation (HPR)	1 / 4 Charging or SI trains taking suction from 1 / 2 LPSI trains with successful switchover to sump (operator action)		
Circle Affected Functions	**Recovery of Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 SLOCA - EIHP (8)		**Fire induced SLOCA - no EIHP available - no mitigation credit**	**WHITE**
2 SLOCA - AFW - PCS - FB (7)			
3 SLOCA - HPR (2,4,6)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 5.6 - Risk Significance Estimation Matrix

Initiating Event Likelihood	Remaining Mitigation Capability Rating (with Examples)						
	-6	-5	-4	-3	-2	-1	0
	3 diverse trains OR 2 multi-train systems OR 1 train + 1 multi-train system + recovery of failed train	1 train + 1 multi-train system OR 2 diverse trains + recovery of failed train	2 diverse trains OR 1 multi-train system + recovery of failed train	1 train + recovery of failed train OR 1 multi-train system OR Operator action + recovery of failed train	1 train OR Operator action OR Operator action under high stress + recovery of failed train	Recovery of failed train OR Operator action under high stress	none
A	Green	White	Yellow	Red	Red	Red	Red
B	Green	Green	White	Yellow	Red	Red	Red
C	Green	Green	Green	White	Yellow	Red	Red
D	Green	Green	Green	Green	White	Yellow	Red
E	Green	Green	Green	Green	Green	White	Yellow
F	Green	Green	Green	Green	Green	Green	White
G	Green	Green	Green	Green	Green	Green	Green
H	Green	Green	Green	Green	Green	Green	Green

Performance Indicators

General Intent

Process for developing Thresholds

PI Reporting

PI INDICATORS

General Intent

Process for Developing Thresholds

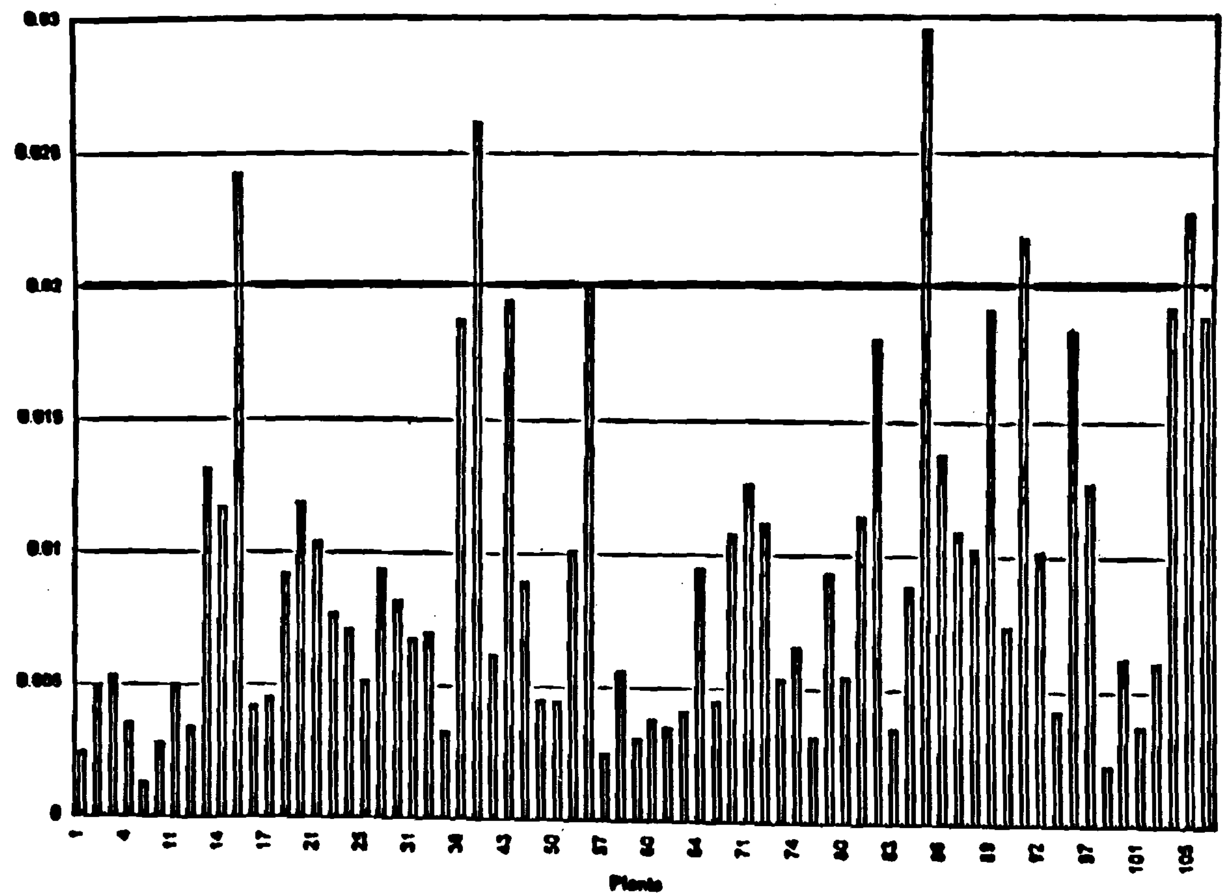
- The Green-White Threshold Was Determined by Experience Data Such That Only a Few Outlier Values Would Be in the White.

- The White-Yellow Threshold Was Established At a Value Such That a Change of the PI Would Result in a Δ CDF >1E-05.

- The Yellow-Red Threshold Was Established Such That a Change in PI Value Greater than the Threshold Would Result in a Δ CDF > 1E-04.

- The Thresholds Were Established Using a Set of PRA Models with the Lower Threshold from the Set Being Used.

PWR AFW UNAVAILABILITY



PI REPORTING

EXAMPLE: EDG SYSTEM UNAVAILABILITY

- PI Definition (from NEI 99-02)

$$\text{Train unavailability} = \frac{\text{unavailable hours (planned + unplanned + fault exposure)}}{\text{hours train required}}$$

$$\text{System unavailability} = \frac{\sum \text{Train unavailability}}{\text{no. of trains}}$$

- Licensee collects unavailable hours for each EDG in latest quarter from logs, equipment histories, etc.

- Licensee submits data to NRC via email

- NRC system automatically calculates PI value per above algorithm

SELECTED ISSUES

- Threshold for the "Green" band
- Results of cross-cutting issue inspections (corrective action)
- Defense in depth aspects of fire protection

Cross Cutting Issues and Initial Implementation of the ROP

Fundamental Assumption of ROP - Cross Cutting Issues Will Be Detected via Baseline Inspection or Performance Indicators

Initial Implementation shows strong tie between plants with weak corrective action programs and plants that move out of licensee response band -(IP 2, Kewaunee, Millstone, Cooper)

No significant precursors caused by cross cutting issues

No additional cross cutting issues identified that require special treatment

Cross Cutting Issues and Initial Implementation of the ROP

Additional actions planned to evaluate adequacy of ROP with regard to cross cutting issues

Review of ASP events and yellow and red inspection findings

Review of inspection reports

Performance metrics to evaluate plants that jump two or more action matrix
columns

Assessment of ROP engagement at plants that reach the degraded cornerstone column of the action matrix