03-4313

UNITED STATES COURT OF APPEALS for the SECOND CIRCUIT

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Riverkeeper, Inc.,

Petitioner,

- against -

SAMUEL J. COLLINS, Director, Office of Nuclear Reactor Regulation; DR. WILLIAM TRAVERS, Executive Director for Operations of the Nuclear REGULATORY COMMISSION; the UNITED STATES OF AMERICA; ENTERGY NUCLEAR INDIAN POINT 2 LLC; ENTERGY NUCLEAR INDIAN POINT 3, LLC; and ENTERGY NUCLEAR OPERATIONS, INC.

Respondents.

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On Petition for Review of a Decision of the Nuclear Regulatory Commission

Joint Appendix: Volume 1 of 3

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IN THE UNITED STATES COURT OF APPEALS
FOR THE SECOND CIRCUIT

RIVERKEEPER, INC., Petitioner,	)	
v.	)	Docket No. 03-4313
COLLINS, et al Respondents	)	

CERTIFIED INDEX OF THE RECORD

I hereby certify that the documents listed and described below in the Certified Index of Record constitute the record of the decision by the U.S. Nuclear Regulatory Commission, the respondent herein, in the above-captioned case.

Respectfully submitted,

Annette L. Vietti-Cook
Secretary of the Commission

Dated at Rockville, Maryland
this 15th day of May, 2003.

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Thursday, May 15, 2003

RAS ID NO.	DOCKET NUMBER	PLANT NAME	DESCRIPTION	DOCUMENT DATE	DOCKET DATE	RELATED CORRES.	ACCESSION NO.
5001	50-003/247/286 (2.206)	INDIAN POINT 1, 2 AND 3	DD-02-06 -- LETTER FROM SAMUEL J. COLLINS TO ALEX MATTHIESSEN ENCLOSING DIRECTOR'S DECISION (ADAMS PACKAGE)	11/18/2002	11/19/2002		ML022630294
5101	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	DD-02-06 -- LETTER SECY TO MATTHIESSEN INFORMING COMMISSION DECLINED ANY REVIEW; DECISION BECAME FINAL AGENCY ACTION ON 12/13/02 - SERVED	12/18/2002	12/18/2002		ML023520485
6391	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	SECTION 2.206 REQUEST FOR EMERGENCY SHUTDOWN OF INDIAN POINT UNITS 2 AND 3, DATED NOVEMBER 8, 2001	11/01/2001	05/13/2003		ML013480179
6392	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	U.S. CONFERENCE OF MAYORS, "IF U.S. CITY/COUNTY METRO ECONOMIES WERE NATIONS," AT HTTP:..WWW.USMAYORS.ORG/CITIES DRIVETHEECONOMY/CHART2_DECAD E.PDF (LAST VISITED NOV. 7, 2001). PETITIONER'S EXHIBIT 1		05/13/2003		
6393	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	DANIEL RUBIN, "NUCLEAR TERRORISM THREAT GROWING," (KNIGHT RIDDER FOREIGN SERVICE, NOVEMBER 3, 2001). PETITIONER'S EXHIBIT 2	11/03/2001	05/13/2003		

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6394	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	CNN, "THREAT AT THREE MILE ISLAND CLOSES AIRPORTS," (OCTOBER 18, 2001). PETITIONER'S EXHIBIT 3	10/18/2001	05/13/2003		
6395	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	INTERNATIONAL ATOMIC ENERGY AGENCY, "CALCULATNG THE NEW GLOBAL NUCLEAR TERRORISM THREAT" (NOVEMBER 1, 2001). PETITIONER'S EXHIBIT 4	11/01/2001	05/13/2003		
6396	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	ERIC PIANIN AND DAN EGGEN, "PREPARATIONS STEPPED UP FOR POSSIBLE NEW ATTACKS," (WASHINGTON POST, NOVEMBER 1, 2001). PETITIONER'S EXHIBIT 5	11/01/2001	05/13/2003		
6397	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	LARA MANSNERUS AND JUDITH MILLER, "TERRORIST DETAILS HIS TRAINING IN AFGHANISTAN," (N.Y. TIMES, JULY 4, 2001). PETITIONER'S EXHIBIT 6	07/04/2001	05/13/2003		
6398	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	NICHOLAS RUFFORD, DAVID LEPPARD AND PAUL EDDY, "NUCLEAR MYSTERY: CRASHED PLANE'S TARGET MAY HAVE BEEN REACTOR," (SUNDAY TIMES: LONDON, UK, OCTOBER 21, 2001). PETITIONER'S EXHIBIT 7	10/21/2001	05/13/2003		

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6399	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	CNN, "FAA RESTRICTS FLIGHTS NEAR WORLD SERIES, NUCLEAR PLANTS" (OCTOBER 30, 2001). PETITONER'S EXHIBIT 8	10/30/2001	05/13/2003		
6400	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	WILLIAM BRANIGIN, "IN AFGHAN JAIL, A TERRORIST WHO WON'T SURRENDER," (WASHINGTON POST, OCTOBER 30, 2001). PETITIONER'S EXHIBIT 9	10/30/2001	05/13/2003		
6401	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	NRC, PRESS RELEASE NO. 01-112, "NRC REACTS TO TERRORIST ATTACKS" (SEPTEMBER 21, 2001). PETITIONER'S EXHIBIT 10	09/21/2001	05/13/2003		
6402	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	NRC REPORT, "TECHNICAL STUDY OF SPENT FUEL POOL ACCIDENT RISK AT DECOMMISSIONING NUCLEAR POWER PLANTS," (OCTOBER 2000). PETITIONER'S EXHIBIT 11	10/31/2001	05/13/2003		
6403	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	US DEPARTMENT OF ENERGY, "HEALTH AND ENVIRONMENTAL CONSEQUENCES OF THE CHERNOBYL NUCLEAR POWER PLANT ACCIDENT," DOE/ER-0332 (JUNE 1987)	06/30/1987	05/13/2003		

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6404	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	V. L. SAILER ET AL. "SEVERE ACCIDENT IN SPENT FUEL POOLS IN SUPPORT OF GENERIC SAFETY ISSUES 82," NUREG/CR-4982 (JULY 1987)	07/31/1987	05/13/2003		
6406	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	ROGER WITHERSPOON, "INDIAN POINT CHIEF: PLANT SAFE FROM POSSIBLE ATTACK," (JOURNAL NEWS, OCTOBER 20, 2001). PETITIONERS' EXHIBIT 12	10/20/2001	05/13/2003		
6407	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	VIBEKE LAROI, "NUCLEAR PLANT SCARE ADDS TO U.S. SECURITY JITTERS," (REUTERS, OCTOBER 18, 2001). PETITIONERS' EXHIBIT 13	10/18/2001	05/13/2003		
6408	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	DOUGLAS PASTERNAK, "A NUCLEAR NIGHTMARE," (U.S. NEWS & WORLD REPORT, SEPTEMBER 17, 2001). PETITONER'S EXHIBIT 14	09/17/2001	05/13/2003		
6409	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	PAUL L. LEVENTHAL, COMMENCEMENT ADDRESS TO THE CLASS OF 2001 AT FRANKLIN & MARSHALL COLLEGE (MAY 13, 2001). PETITIONER'S EXHIBIT 15	05/13/2001	05/13/2003		

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6410	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	SHAWN COHEN, "NRC FLAGS ConEd WITH RED TAG FOR INDIAN POINT," (JOURNAL NEWS, NOVEMBER 21, 2000). PETITIONER'S EXHIBIT 16	11/21/2000	05/13/2003		
6411	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	SANDIA LABS, NRC, "CALCULATION OF REACTOR ACCIDENT CONSEQUENCES" (1982) (CRAC-2 REPORT)	11/15/1982	05/13/2003		
6412	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	"AT INDIAN PT., A HISTORY OF NUCLEAR POWER, PROBLEMS AND CONTROVERSY," (N.Y. TIMES, MAY 6, 1983). PETITIONER'S EXHIBIT 17	05/06/1983	05/13/2003		
6414	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	DECLARATION OF 20 DECEMBER 2001 BY GORDON THOMPSON IN SUPPORT OF A PETITION BY RIVERKEEPER, INC., DATED DECEMBER 20, 2001.	12/20/2001	05/13/2003		ML013610467
6415	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	LETTER FROM SAMUEL COLLINS (NRC) TO ALEX MATTHIESSEN (RIVERKEEPER) ACKNOWLEDGING RECEIPT OF RIVERKEEPER'S 2.206 PETITION AND DENYING IMMEDIATE CLOSURE OF INDIAN POINT 2 AND 3, DATED DECEMBER 20, 2001.	12/20/2001	05/13/2003		ML013510656

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6416	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	LICENSEE'S RESPONSE TO RIVERKEPER, INC.'S SECTION 2.206 REQUEST FOR EMERGENCY SHUTDOWN OF INDIAN POINT UNITS 2 AND 3, DATED FEBRUARY 11, 2002	02/11/2002	05/13/2003		ML020430500
6417	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	NRC SUPPLEMENTAL INSPECTION REPORT NO. 05000247/2001-002, INDIAN POINT UNIT 2 (APRIL 10, 2001)	04/10/2001	05/13/2003		ML011000373
6418	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	AEC REGULATORY GUIDE 1.78 (JUNE 1974)	06/30/1974	05/13/2003		ML003740298
6419	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	NRC INSPECTION MANUAL, INSPECTION PROCEDURE 81110, OPERATIONAL SAFEGUARDS RESPONSE EVALUATION (JULY 1, 1997)	07/01/1997	05/13/2003		
6420	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	LETTER FROM NRC CHAIRMAN RICHARD MESERVE TO SENATOR JAMES JEFFORDS (DECEMBER 17, 2001)	12/17/2001	05/13/2003		ML013380256

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6421	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	LETTER FROM G. SCOTT BARBER (NRC) TO J. BARRETT (ENTERGY) (DECEMBER 31, 2001) (ENCLOSING NRC INSPECTION REPORT 50/286/01-09)	12/31/2001	05/13/2003		ML013650345
6422	50-003/247/286 (2.206)	INDIAN POINT 1 AND 2	NRC OFFICE OF NUCLEAR REACTOR REGULATION, DIRECTIVE 8.11, "REVIEW PROCESS FOR 10 C.F.R. 2.206 PETITIONS" (OCTOBER 25, 2000)	10/25/2000	05/13/2003		ML003670845
6423	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	"MALEVOLENT USE OF VEHICLES AT NUCLEAR POWER PLANTS" 59 FED. REG. 38889 (1994)	08/01/1994	05/13/2003		
6424	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	"FINAL RULE: PHYSICAL PROTECTION FOR SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE" 63 FED. REG. 26955 (1998)	05/15/1998	05/13/2003		
6425	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	NUREG-0800, SECTION 3.5.1.6, "STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS"		05/13/2003		

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6426	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	NUREG-0654/FEMA-REP-1 (REV. 1), "CRITERIA FOR PREPARATION AND EVALUATION OF RADIOLOGICAL EMERGENCY RESPONSE PLANS AND PREPAREDNESS IN SUPPORT OF NUCLEAR POWER PLANTS." AVAILABLE ON USNRC ELECTRONIC READING ROOM		05/13/2003		
6427	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	NRC INSPECTION REPORT 95000286/2000-10 (DECEMBER 8, 2000)	12/08/2000	05/13/2003		ML003775738
6428	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	"EVALUATION OF AIRCRAFT-CRASH-HAZARDS ANALYSES FOR NUCLEAR POWER PLANTS." NUREG/CR-2859 (JUNE 1982)	06/30/1982	05/13/2003		
6429	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	PROPOSED DIRECTOR'S DECISION UNDER 10 CFR 2.206, DATED MAY 16, 2002	05/16/2002	05/13/2003		ML021340261
6430	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	RIVERKEEPER'S COMMENTS ON MAY 16, 2002 PROPOSED DIRECTOR'S DECISION ON RIVERKEEPER'S NOVEMBER 8TH PETITION 2.206 REQUEST FOR EMERGENCY SHUTDOWN OF INDIAN POINT UNITS 2 AND 3, DATED AUGUST 9, 2002	08/09/2002	05/13/2003		ML022630472

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6431	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	NATIONAL RESEARCH COUNCIL, "MAKING THE NATION SAFER: THE ROLE OF SCIENCE AND TECHNOLOGY IN COUNTERING TERRORISM," (2002), PETITIONER'S EXHIBIT A		05/13/2003		
6433	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	ASSOCIATED PRESS, "PLANES OFTEN ENTER PROHIBITED AIR," (APRIL 5, 2002). PETITIONER'S EXHIBIT B	04/05/2002	05/13/2003		
6434	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	ABC NEWS, "SUBWAYS, SCUBA DIVERS AND LANDMARKS," (MAY 27, 2002). PETITIONER'S EXHIBIT C	05/27/2002	05/13/2003		
6435	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	CBS NEWS, "DOZENS OF AIRPORT WORKERS ARRESTED," (APRIL 23, 2002). PETITIONER'S EXHIBIT D	04/23/2002	05/13/2003		
6436	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	CBS NEWS, "AIRPORT SECURITY GETS AN 'F'" (MARCH 25, 2002). PETITIONER'S EXHIBIT E	03/25/2002	05/13/2003		

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6437	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	NUCLEAR CONTROL INSTITUTE, "THE IMPACT OF NUCLEAR PLANT SHUTDOWN ON SEVERE ACCIDENT CONSEQUENCES," (FEBRUARY 12, 2002)	02/12/2002	05/13/2003		
6438	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	REP. EDWARD MARKEY, SECURITY GAP: "A HARD LOOK AT THE SOFT SPOTS IN OUR CIVILIAN NUCLEAR REACTOR SECURITY," (MARCH 25, 2002)	03/25/2002	05/13/2003		
6439	50-003/247/286 (2.206)	INDIAN POINT 1, 2, AND 3	NRC, REGULATORY ISSUES SUMMARY (RIS) 2001-16, "UPDATE OF EVACUATION TIME ESTIMATES" (AUGUST 1, 2001)	08/01/2001	05/13/2003		ML012070310

November 18, 2002

Mr. Alex Matthiessen
Executive Director
Riverkeeper, Inc.
25 Wing & Wing
Garrison, NY 10524

Dear Mr. Matthiessen:

This letter responds to the petition you filed with Dr. William Travers, Executive Director for Operations, of the U.S. Nuclear Regulatory Commission (NRC) pursuant to Section 2.206 of Title 10 of the *Code of Federal Regulations* (10 CFR 2.206) on November 8, 2001, and as supplemented on December 20, 2001. In your petition, you requested that the NRC: (1) order the licensee to suspend operations, revoke the operating license, or adopt other measures resulting in a temporary shutdown of the Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and 3); (2) order the licensee to conduct a full review of the facility's vulnerabilities, security measures and evacuation plans; (3) require the licensee to provide information documenting the existing and readily attainable security measures which protect the IP facility against land, water, and airborne terrorist attacks; (4) immediately modify the IP2 and 3 operating licenses to mandate certain specified security measures sufficient to protect the facility; and (5) order the revision of the licensee's Emergency Response Plan and Westchester County's Radiological Emergency Response Plan (RERP) to account for possible terrorist attacks and prepare a comprehensive response to multiple, simultaneous attacks in the region which may impair the efficient evacuation of the area. In addition, you stated that if, after conducting a full review of the facility's vulnerabilities, security measures, and evacuation plans, the NRC finds that it cannot sufficiently ensure the security of the IP facility against terrorist threats, the NRC should take prompt action to permanently retire the facility. Further, separately from the above issues, you requested that the NRC order the licensee to undertake the immediate conversion of the current spent fuel storage technology from a water-cooled system to a dry cask system.

On December 20, 2001, the NRC staff acknowledged receiving your petition and stated that, pursuant to 10 CFR 2.206, your petition would be acted upon within a reasonable time. You were also told that the NRC did not consider the immediate closure of IP2 and 3 to be necessary to provide adequate protection for the public's health and safety in light of the defense-in-depth concept incorporated into the facility's design and the heightened security measures implemented in response to the events of September 11, 2001.

Although the NRC staff did not request Entergy Nuclear Operations, Inc. (ENO), to provide information related to the petition, ENO responded on February 11, 2002, and the information provided was considered by the staff in its evaluation of the petition.

The staff sent a copy of the Proposed Director's Decision to you and to ENO for comment on May 16, 2002. You responded with comments on August 9, 2002. ENO did not provide any comments. Your comments and the staff's response to them are included as enclosures.

With regard to the issues raised by the Petitioners, the NRC has, in effect, partially granted the Petitioners' request for an immediate security upgrade at IP2 and 3. As stated in its letter to the Petitioners on December 20, 2001, the NRC took action to enhance security at all nuclear facilities, including IP2 and 3, on September 11, 2001. Immediately after the attacks, the NRC advised all nuclear power plants to go to the highest level of security, which they promptly did. These facilities have remained at a heightened security level since that time. Additionally, the NRC issued Orders to all operating commercial nuclear power plants on February 25, 2002, to implement interim compensatory security measures for the current threat environment. Some of the requirements formalized a series of security measures that NRC licensees had taken in response to advisories issued by the NRC, and others were security enhancements which have emerged from the Commission's ongoing comprehensive security review. In general, the requirements include increased patrols, augmented security forces and capabilities, additional security posts, installation of additional physical barriers, vehicle checks at greater stand-off distances, enhanced coordination with law enforcement and military authorities, and more restrictive site access controls for all personnel. The NRC continues to work with other Federal agencies and is monitoring relevant information it receives on security matters at nuclear facilities. The NRC is prepared to make immediate adjustments as necessary to ensure adequate protection of the public's health and safety. On the basis of these actions, the Petitioners' request that the licensee conduct a full review of the facility's vulnerabilities, security measures, and evacuation plans has been, in effect, partially granted. Regarding the Petitioners' request for specific information about the security measures, the NRC's policy is to not release safeguards information to the public. Thus, this request is denied.

The NRC in its February 25, 2002, Orders also directed licensees to evaluate and address potential vulnerabilities to maintain or restore cooling to the core, containment, and spent fuel pool and to develop specific guidance and strategies to respond to an event that damages large areas of the plant due to explosions or fires. These strategies are intended to help licensees to identify and utilize any remaining onsite or offsite equipment and capabilities. If NRC's ongoing security review recommends any other security measures, the NRC will take appropriate action.

The NRC denies the Petitioners' request to mandate certain security measures, as specified by the Petitioners, for the protection of the facility, such as a system to defend a no-fly zone. As part of its ongoing comprehensive security review, the NRC is examining the threat environment in coordination with other Federal agencies and the use of governmental assets to augment the licensee's response. These organizations will define the appropriate boundary between the public and private sector in the defense of nuclear facilities. Further, the current security requirements, along with the enhancements in the February 25 Orders, provide reasonable assurance of the protection of the facility.

The NRC finds that the existing emergency response plans are flexible enough to respond to a wide variety of adverse conditions, including a terrorist attack. The NRC advisories and the Orders issued since September 11, 2001, directed licensees to take specific actions deemed appropriate to ensure continued improvements to existing emergency response plans. The Petitioners' concern that the emergency plans do not contemplate multiple attacks on the infrastructure is alleviated by the fact that the emergency plans are intended to be broad and flexible enough to respond to a wide spectrum of events. Thus, the Petitioners' request that

the onsite and offsite emergency plans be revised to account for possible terrorist attacks has been, in part, granted.

The NRC found that the current spent fuel storage system and the security provisions at IP adequately protect the spent fuel. Thus, the Petitioners' request to order the installation of a dry-cask storage facility is denied. However, the licensee has stated its intention to add such a facility.

A copy of the Director's Decision (DD-02-06) will be filed with the Secretary of the Commission for the Commission to review in accordance with 10 CFR 2.206(c). As provided for by this regulation, the decision will constitute the final action of the Commission 25 days after the date of the decision unless the Commission, on its own motion, institutes a review of the decision within that time. The documents cited in the enclosed decision are available in the Agencywide Documents Access and Management System (ADAMS) for inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, and from the ADAMS Public Library component on the NRC Web site, http://www.nrc.gov/reading-rm.html (the Public Electronic Reading Room).

The incoming petition was originally withheld from the public document room due to the potential for sensitive, security-related information to be included. When the NRC received your letter, the criteria for releasing security-related information was still being determined in light of the events of September 11, 2001. Your incoming letter, and subsequent correspondence, were later made publicly available.

I have also enclosed a copy of the notice of "Issuance of Director's Decision Under 10 CFR 2.206" that has been filed with the Office of the Federal Register for publication.

Please feel free to contact the petition manager, Patrick Milano, at 301-415-1457 to discuss any questions related to this petition. I thank you for your time and interest in nuclear power plant security.

Sincerely,

/RA/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Docket Nos. 50-003, 50-247, and 50-286

Enclosures:
1. Director's Decision 02-06
2. Petitioners' Comments on Proposed Director's Decision
3. Staff's Response to Petitioners' Comments
4. *Federal Register* Notice

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7590-01-P

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-003, 50-247, AND 50-286

LICENSE NOS. DPR-5, DPR-26, AND DPR-64

ENTERGY NUCLEAR OPERATIONS, INC.

NOTICE OF ISSUANCE OF DIRECTOR'S DECISION UNDER 10 CFR 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation, has issued a Director's Decision with regard to a petition dated November 8, 2001, filed by Riverkeeper, Inc., et al., hereinafter referred to as the "Petitioners." The petition was supplemented on December 20, 2001. The petition concerns the operation of the Indian Point Nuclear Generating Unit Nos. 1, 2, and 3 (IP1, 2, and 3).

The petition requested that the U.S. Nuclear Regulatory Commission (NRC): (1) order the licensee to suspend operations, revoke the operating license, or adopt other measures resulting in a temporary shutdown of IP2 and 3; (2) order the licensee to conduct a full review of the facility's vulnerabilities, security measures, and evacuation plans; (3) require the licensee to provide information documenting the existing and readily attainable security measures which protect the IP facility against land, water, and airborne terrorist attacks; (4) immediately modify the IP2 and 3 operating licenses to mandate certain specified security measures sufficient to protect the facility; and (5) order the revision of the licensee's emergency response plan and Westchester County's radiological emergency response plan (RERP) to account for possible terrorist attacks and prepare a comprehensive response to multiple, simultaneous attacks in the region, which could impair the efficient evacuation of the area. In addition, the Petitioners requested that the NRC take prompt action to permanently retire the facility if, after conducting a full review of the facility's vulnerabilities, security

measures, and evacuation plans, the NRC finds that the IP facility cannot be adequately protected against terrorist threats. Further, separately from the above issues, the Petitioners requested that the NRC order the licensee to undertake the immediate conversion of the current water-cooled spent fuel storage system to a dry-cask system.

As the basis for the November 8, 2001, request, the Petitioners stated that: (1) the IP facility is a plausible target of future terrorist actions, (2) actual threats against nuclear power plants have been documented, (3) IP is currently vulnerable to a catastrophic terrorist attack, (4) a terrorist attack on IP2 and 3 would have significant public health, environmental, and economic impacts, and (5) the Westchester County's RERP is inadequate because it is based on erroneous assumptions.

The NRC sent a copy of the proposed Director's Decision to the Petitioners and to the licensee for comment on May 16, 2002. The Petitioners responded with comments on August 9, 2002, and the licensee had no comments. The Petitioners' comments and the NRC staff's response to them are included with the Director's Decision.

The Director of the Office of Nuclear Reactor Regulation has determined that the request to order the licensee to suspend operations, revoke the operating license, or adopt other measures resulting in a temporary shutdown of IP2 and 3, be denied. The reasons for this decision, along with the reasons for decisions regarding the remaining Petitioners' requests, are explained in the Director's Decision pursuant to 10 CFR 2.206 (DD 02-06), the complete text of which is available in the Agencywide Documents Access and Management System (ADAMS) for inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, and from the NRC Web site (http://www.nrc.gov/reading-rm.html).

As stated in its letter to the Petitioners on December 20, 2001, the NRC has, in effect, partially granted the Petitioners' request for an immediate security upgrade at IP2 and 3. On September 11, 2001, the NRC took action to enhance security at all nuclear facilities, including IP2 and 3. Immediately after the attacks, the NRC advised all nuclear power plants to go to the highest level of security, which they promptly did. These facilities have remained at a heightened security level since that time. The NRC continues to work with other Federal agencies and is monitoring relevant information it receives on security matters at nuclear facilities. The NRC is prepared to make immediate adjustments as necessary to ensure adequate protection of the public.

The NRC issued Orders on February 25, 2002, to all commercial nuclear power plants to implement interim compensatory security measures for the current threat environment. Some of the requirements made mandatory by the Orders formalized the security measures that NRC licensees had taken in response to advisories issued by the NRC in the aftermath of the September 11 terrorist attacks. The Orders also imposed additional security enhancements, which have emerged based on the NRC's assessment of the current threat environment and its ongoing security review. The requirements will remain in effect until the NRC determines that the level of threat has diminished, or that other security changes are needed. The specific actions are sensitive, but include increased patrols, augmented security forces and capabilities, additional security posts, installation of additional physical barriers, vehicle checks at greater stand-off distances, enhanced coordination with law enforcement and military authorities and more restrictive site access controls for all personnel. Regarding the Petitioners' request for specific information about the security measures, the NRC's policy is to not release safeguards information to the public. Thus, this request is denied.

The NRC in its February 25, 2002, Orders also directed licensees to evaluate and address potential vulnerabilities to maintain or restore cooling to the core, containment, and spent fuel pool and to develop specific guidance and strategies to respond to an event that damages large areas of the plant due to explosions or fires. These strategies are intended to help licensees to identify and utilize any remaining onsite or offsite equipment and capabilities. If NRC's ongoing security review recommends any other security measures, the NRC will take appropriate action.

The NRC denies the Petitioners' request to mandate certain security measures, as specified by the Petitioners, for the protection of the facility, such as a system to defend a no-fly zone. The NRC considers that the collective measures taken since September 11, 2001, provide adequate protection of public health and safety.

The NRC finds that the existing emergency response plans are flexible enough to respond to a wide variety of adverse conditions, including a terrorist attack. The NRC advisories and the Orders issued since September 11, 2001, directed licensees to take specific actions deemed appropriate to ensure continued improvements to existing emergency response plans. The Petitioners' concern that the emergency plans do not contemplate multiple attacks on the infrastructure is alleviated by the fact that the emergency plans are intended to be broad and flexible enough to respond to a wide spectrum of events. Thus, the Petitioners' request that the onsite and offsite emergency plans be revised to account for possible terrorist attacks has been, in part, granted.

The NRC finds that the current spent fuel storage system and the security provisions at IP adequately protect the spent fuel. Thus, the Petitioners' request to order the installation of a dry-cask storage facility is denied. However, the licensee has stated its intention to add such a facility.

A copy of the Director's Decision will be filed with the Secretary of the Commission for the Commission's review in accordance with 10 CFR 2.206 of the Commission's regulations. As provided for by this regulation, the Director's Decision will constitute the final action of the

Commission 25 days after the date of the decision, unless the Commission, on its own motion, institutes a review of the Director's Decision in that time.

Dated at Rockville, Maryland, this 18th day of November 2002.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

DD-02-06

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

Samuel J. Collins, Director

In the Matter of	)	Docket Nos. 50-003, 50-247, and 50-286
ENTERGY NUCLEAR OPERATIONS, INC.	)	License Nos. DPR-5, DPR-26, and DPR-64
(Indian Point Nuclear Generating Unit Nos. 1, 2, and 3)	)	(10 CFR 2.206)

DIRECTOR'S DECISION UNDER 10 CFR 2.206

I. Introduction

By letter dated November 8, 2001, as supplemented on December 20, 2001, Riverkeeper, Inc., et al. filed a Petition pursuant to Title 10 of the *Code of Federal Regulations*, Section 2.206 (10 CFR 2.206). The Petitioners requested that the U.S. Nuclear Regulatory Commission (NRC) take the following actions: (1) order the licensee to suspend operations, revoke the operating license, or adopt other measures resulting in a temporary shutdown of the Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and 3); (2) order the licensee to conduct a full review of the facility's vulnerabilities, security measures, and evacuation plans; (3) require the licensee to provide information documenting the existing and readily attainable security measures which protect the IP facility against land, water, and airborne terrorist attacks; (4) immediately modify the IP2 and 3 operating licenses to mandate certain specified security measures sufficient to protect the facility; and (5) order the revision of the licensee's emergency response plan and Westchester County's Radiological Emergency Response Plan (RERP) to account for possible terrorist attacks and prepare a comprehensive response to multiple, simultaneous attacks in the region, which could impair the efficient evacuation of the

area. In addition, the Petitioners requested that the NRC take prompt action to permanently retire the facility if, after conducting a full review of the facility's vulnerabilities, security measures, and evacuation plans, the NRC finds that the IP facility cannot be adequately protected against terrorist threats. Further, separately from the above issues, the Petitioners requested that the NRC order the licensee to undertake the immediate conversion of the current water-cooled spent fuel storage system to a dry cask system. The bases for the requests are that (1) the IP facility is a plausible target of future terrorist actions, (2) actual threats against nuclear power plants have been documented, (3) IP is currently vulnerable to a catastrophic terrorist attack, (4) a terrorist attack on IP2 and 3 would have significant public health, environmental, and economic impacts, and (5) the Westchester County's RERP is inadequate because it is based on erroneous assumptions.

In a letter dated December 20, 2001, the NRC informed the Petitioners that their request for a full review of the facility's vulnerabilities, security measures, and evacuation plans was, in effect, partially granted, because the NRC had already taken action to require licensees to enhance security and the Commission had directed the staff to undertake a comprehensive review of plant security. In light of the defense-in-depth concept incorporated into the facility's design and the heightened security measures implemented in response to the events of September 11, 2001, the NRC did not consider the immediate closure of IP2 and 3 to be necessary to provide adequate protection of the public health and safety.

In its December 20, 2001, letter, the NRC told the Petitioners that a public meeting or telephone conference with the NRR Petition Review Board was not necessary or appropriate at the time since the Petitioners' request was already being treated as a 2.206 Petition and because of the possible sensitive nature of the information. Under normal circumstances, the NRC would follow Management Directive (MD) 8.11, "Review Process for 10 CFR 2.206

Petitions," when reviewing requests for enforcement action; however, since the Petition involved possible sensitive information, the NRC deferred application of certain public aspects of the MD 8.11 process pending further developments of the NRC's security review.

On December 20, 2001, the Petitioners provided a declaration from Dr. Gordon Thompson dated December 20, 2001, and requested that the declaration be included as a supplement to their Petition. The NRC treated the declaration as a supplement to the Petition. Although the NRC had initially withheld the Petition from public distribution pending Commission guidance about public dissemination of potential security information, the NRC has now determined that the Petition can be made publicly available. Therefore, the documents are available in the NRC's Agencywide Documents Access and Management System (ADAMS) for inspection at the Commission's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records are also accessible from the ADAMS Public Electronic Reading Room on the NRC Web site http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or have problems in accessing the documents located in ADAMS should contact the NRC PDR reference staff by telephone at 1-800-397-4209 or 301-415-4737 or by e-mail to pdr@nrc.gov.

Entergy Nuclear Operations, Inc. (the licensee), responded to the Petition on February 11, 2002, and the staff considered the information in reviewing the Petition.

The NRC sent a copy of the proposed Director's Decision to the Petitioners and to the licensee for comment on May 16, 2002. The Petitioners responded with comments on August 9, 2002. The licensee did not provide comments. The comments and the NRC staff's response to them are enclosed with the final Director's Decision.

II. Discussion

Full Review of Vulnerabilities and Security Measures

In the Petition, as supplemented, the Petitioners requested that the NRC order the licensee to conduct a full review of the facility's vulnerabilities and security measures. The Petitioners stated that the reactor, spent fuel, control rooms, and electrical switching were vulnerable to terrorist attack. The Petitioners' request was based on the following assertions: (1) IP2 and 3 are a plausible target because of the population density of the surrounding area and the proximity to New York City, (2) news releases have documented threats against nuclear facilities, (3) an operational plant is more vulnerable, (4) an attack could damage cooling to the spent fuel pools (SFPs) and/or drain the pools, leading to fuel cladding oxidation, fire, and release of radioactive materials, and (5) the design-basis threat did not consider a terrorist attack. The Petitioners also stated that the facility is not currently equipped to defend itself from terrorist attacks, the licensee has a poor record in security and emergency preparedness, and nuclear industry security forces have repeatedly failed to repel mock attacks. The Petitioners also believe that an attack on an operating reactor would force plant operators to face competing interests between safe operations and physical security.

Staff's Response

The Petitioners' request for a review of vulnerabilities and security measures has been partially granted based on actions initiated by the NRC following the events of September 11, 2001. The NRC concludes that Indian Point has sufficient security measures in place to defend itself from a broad spectrum of potential terrorist attacks. The basis for these conclusions is discussed below.

The NRC and its licensees have dealt with the issue of protection of licensed facilities against sabotage or attack for a number of years. Security against sabotage has been an

important part of the NRC's regulatory activities, with defense-in-depth as the guiding design and operating principle. NRC regulations ensure that nuclear power plants are among the most hardened and secure industrial facilities in our nation. The many layers of protection offered by robust plant design features, sophisticated surveillance equipment, physical security protective features, professional security forces, access authorization requirements, and NRC regulatory oversight provide an effective deterrence against potential terrorist activities that could target equipment vital to nuclear safety.

The NRC requirements for the defense of nuclear power plants are defined, in part, by the "design basis threat" (DBT). The DBT is specified in general terms in 10 CFR 73.1 and in greater detail in sensitive documents. The DBT was prepared by safeguards experts on the basis of information from the Department of Energy and the intelligence community about terrorist-related information both abroad and in the United States. The DBT is a reasonable characterization of an adversary force against which nuclear power plant licensees must design their physical protection systems and response strategies.

In 10 CFR Part 73, "Physical Protection of Plants and Materials," the NRC provides detailed requirements designed to protect nuclear power plants against acts of radiological sabotage, prevent the theft of special nuclear material, and protect safeguards information against unauthorized release. The requirements of Part 73 are summarized as follows:

1. The licensee permits only authorized activities and conditions within established protected areas, material access areas, and vital areas by using controls and procedures, defined boundaries, detection, communication and surveillance subsystems, and by establishing schedules of authorized operations.
2. The licensee prevents unauthorized access of persons, vehicles and objects into protected and vital areas by using detection and barrier systems.

3. The licensee provides for authorized access and assures detection of and response to unauthorized penetrations of the protected area.
4. The licensee permits only authorized control and movement of special nuclear material.
5. The licensee provides response capabilities to assure that NRC requirements are achieved.
6. The licensee maintains a well-equipped and highly trained security organization.
7. The licensee installs physical barriers to protect vital equipment and material.
8. The licensee installs detection, surveillance, and alarm systems capable of sensing unauthorized penetrations of isolation zones and ensuring a prompt response action.
9. The licensee provides access authorization (e.g., background checks, routine worker screening, badging, etc.) programs and procedures.
10. The licensee ensures that all guards and armed response individuals have the ability to communicate with a continuously manned alarm station.
11. The licensee establishes an effective testing and maintenance program to verify that all physical barriers, and detection and alarm systems are capable of meeting NRC requirements.

Licensees are also required to develop specific physical security plans (PSPs) and submit these plans to the NRC for approval before implementing them. The NRC conducts periodic inspections of the licensees' security programs. Performance testing of physical security has been conducted by the NRC staff through Operational Safeguards Response Evaluations. In addition, the licensees are required to establish a liaison with local law enforcement organizations for added assistance in the event of an attack.

Shortly after September 11, 2001, the NRC recognized the need to reexamine the basic assumptions underlying the current nuclear facility security and safeguards programs.

Chairman Richard A. Meserve, with the full support of the Commission, directed the staff to undertake a comprehensive review of the NRC's security regulations and programs. This is an ongoing review and as results become available, they will be evaluated and, if appropriate, incorporated into NRC's regulatory processes. The review includes consultation with the Office of Homeland Security, the Federal Bureau of Investigation (FBI), the Departments of Transportation and Energy, and others. The NRC's participation with these agencies allows the NRC to communicate its actions to other Federal agencies, ensuring an appropriate and balanced response throughout the nation's entire critical energy infrastructure.

The attacks of September 11, 2001, were unprecedented and required the NRC and its licensees to reevaluate the type of assault that might be mounted against a nuclear power plant. As a result, on February 25, 2002, the NRC issued Orders to all operating power reactor facilities to require that certain interim compensatory security measures be taken beyond those called for by current regulations. Although licensee responses to the prior NRC Threat and Safeguards Advisories provided reasonable assurance of adequate protection of public health and safety, the NRC determined that certain compensatory measures were prudent to address the current threat environment in a consistent manner throughout the nuclear reactor industry. The Orders formalized a series of steps that nuclear power plant licensees had been advised to take by the NRC in the aftermath of the terrorist attacks on September 11 and added certain security enhancements. For security reasons, the details of these interim compensatory measures cannot be made public. Some of the specific measures implemented by the licensees in response to the advisories and interim compensatory measures included increased patrols, augmented security forces and capabilities, additional security posts, installation of additional physical barriers, vehicle checks at greater stand-off distances, enhanced coordination with law enforcement and military authorities and more restrictive site

access controls for all personnel. The Orders also directed licensees to evaluate and address potential vulnerabilities to maintain or restore cooling to the core, containment, and spent fuel pool and to develop specific guidance and strategies to respond to an event resulting in damage to large areas of the plant due to explosions or fires. These strategies are intended to help identify and utilize any remaining equipment and capabilities to maintain or restore core, containment, and spent fuel pool cooling, including both onsite and offsite resources. These requirements will remain in effect until the NRC notifies licensees that the threat environment has significantly changed or until the NRC determines, as a result of the ongoing comprehensive reevaluation of current safeguards and security programs, that other changes are needed.

The Petitioners are correct that the DBT did not consider a terrorist attack such as occurred on September 11, 2001. As part of the comprehensive review of safeguards vulnerabilities, the NRC will reexamine the DBT and modify it as appropriate. As in the past, the NRC will coordinate its evaluation with various other Government agencies and discuss resource commitments with the military, the States, and local law enforcement. If a credible vulnerability is identified that is not addressed by another Federal agency, the NRC staff will consider additional physical protection, material control, and other appropriate requirements.

Although the NRC cannot rule out the possibility of future terrorist activity directed at a licensee's site before implementing any further enhancements to its safeguards programs, the NRC believes that these facilities can continue to operate safely.

The staff also recognizes that design and construction of commercial nuclear power plants could contribute to their survivability in the event of an attack not considered by the current design-basis threat, such as an aircraft impact. Nuclear power plant design is based on defense-in-depth principles, and includes many features to protect public health and safety.

For example, reinforced containment buildings and redundant safety systems would help trained operators prevent or limit the release of radioactive material in the event of a terrorist attack. In addition, NRC requirements for coping with fires and station blackout (loss of offsite and onsite power) provide added capability to bring the plant to safe shutdown conditions assuming such aspects as loss of the control room or failure of the emergency diesel generators.

The NRC requires careful background checks (to minimize the risk of insider assistance) and facility access controls, delay barriers, and intrusion detection systems (to detect potential attackers). The NRC also requires licensees to be able to respond with force to a group of armed attackers, using protective strategies involving layers of defense. Therefore, the NRC believes that the facilities are adequate to withstand many of the challenges from safety or safeguards events, such as armed assaults.

In summary, a robust security program existed at IP prior to the events of September 11, 2001. Since September 11, the NRC has initiated a review of nuclear facility security and safeguards programs, and has taken action to enhance security in the interim.

Full Review of Radiological Emergency Preparedness and Evacuation Planning

In its December 20 supplement, the Petitioners cited a prior NRC study prepared by Sandia National Laboratory that discussed source terms and potential radiological consequences of an attack on IP. The Petitioners were concerned about the economic and environmental consequences of an attack causing a massive release of radioactive materials.

Regarding emergency preparedness planning, the Petitioners believe that the IP onsite and offsite emergency plans did not envision an act of terrorism of the magnitude seen on September 11, 2001. Additionally, the Petitioners stated that the Westchester County RERP is

inadequate and does not consider the possibility of multiple simultaneous attacks on vital infrastructure relied on in the current plan.

Staff's Response

The NRC finds that the emergency preparedness plans and evacuation planning at IP2 and 3 are appropriate to use in response to a radiological emergency, including a release caused by a terrorist attack. The basis for this conclusion is discussed below.

The overall objective of emergency response planning is to minimize the dose to the public for a spectrum of accidents that could produce offsite doses in excess of protective action guidelines. No single accident sequence should be isolated as the one for which to plan because each accident could have different consequences, both in nature and degree. Emergency plans are intended to be broad and flexible enough to respond to a wide spectrum of situations, including various initiating events, sources of release, types of nuclides released, and magnitude, timing, or duration of release.

The NRC and the Federal Emergency Management Agency (FEMA) are the two Federal agencies responsible for evaluating emergency preparedness at and around nuclear power plants. The NRC is responsible for evaluating the adequacy of onsite emergency plans developed by the utility, while FEMA is responsible for assessing the adequacy of offsite (State and local) radiological emergency planning and preparedness activities. The NRC requires licensees to have detailed procedures for responding to events, making timely notifications to appropriate authorities, and providing accurate radiological information. For the offsite plans, the NRC relies on FEMA's findings in determining whether there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The licensee, local and county emergency response officials, and State emergency management officials discuss and agree on the facility's emergency response plan.

NRC regulations require the establishment of a plume exposure pathway emergency planning zone (EPZ) about 10 miles in radius and an ingestion exposure pathway EPZ about 50 miles in radius around each nuclear power plant site.

In the unlikely event of a severe reactor accident with offsite consequences, NRC guidance calls for the prompt evacuation of the population within a 2-mile radius of the plant and about 5 miles in the downwind direction. The guidance states that these protective actions would be expanded, as necessary, based on further assessment of plant conditions, dose assessment, and field monitoring information. At longer distances, shelter is usually the appropriate protective action, followed by relocation of segments of the population, if warranted by the results and analysis of radiological measurements taken in the field. The main protective action planned for the 50-mile EPZ is protection of the public from the ingestion of contaminated food and water. It is considered extremely unlikely that evacuation would be required at a distance of 50 miles even after the most severe accident. The planning established for the 10-mile and 50-mile EPZs, the decreasing consequences and increasing time available for taking protective actions as the distance from the plant increases, and the availability of monitoring data on which to base protective action decisions provide assurance that appropriate protective actions would be taken to protect the population within 50 miles of a site.

NRC regulations also require that the applicant for a nuclear power reactor operating license provide an analysis of the time required to evacuate and take other protective actions within the plume exposure pathway EPZ. This analysis is referred to as the "evacuation time estimate" (ETE). There are no preset minimum evacuation times that a nuclear power plant site must meet. However, the NRC expects that the ETEs for a site are a reasonably accurate reflection of the time it would take to evacuate the site environs under normal and adverse

conditions. ETEs are mostly used to identify potential traffic bottlenecks so that appropriate traffic control plans can be developed. Nuclear power reactor licensees are expected to review and revise their ETEs for their sites. The revisions must take into account changes in population, road capacities, potential traffic impediments, and other factors affecting the ETEs. The ETEs are assessment tools used by decision makers for determining whether evacuation is the preferred protective action option for the general public under specific accident and offsite conditions.

On August 1, 2001, the NRC issued Regulatory Issues Summary (RIS) 2001-16, "Update of Evacuation Time Estimates," to all holders of operating licenses for nuclear power plants. In this RIS, the NRC alerted licensees of the possible need to update ETEs as a result of the 2000 Census. The licensee is currently preparing a new ETE report for IP2 and 3.

FEMA has established the Radiological Emergency Preparedness Program to (1) ensure that the health and safety of citizens living around commercial nuclear power plants can be adequately protected in the event of a nuclear power plant accident, (2) inform and educate the public about radiological emergency preparedness, and (3) make findings and determinations as to the adequacy of State and local plans and the capability of State and local governments to effectively implement these plans and preparedness measures. Federal agencies also have plans in place to coordinate their response activities and share their resources in support of State and local officials during an emergency. Coordination of activities includes joint planning and training sessions and exercise participation. Emergency plans are continually improved based on experience gained through plan implementation and as a result of exercises, drills, and actual events.

In late January 2002, the State of New York issued its annual letter of certification to FEMA. By this letter, the State informed FEMA that specific preparedness activities have been

completed including training and the updating of State and local plans. However, the updating of State and local plans is an ongoing activity. The NRC staff understands that the State and counties have addressed the adequacy of evacuation plans through their required review process in preparation for the exercise conducted in September 2002 and, in doing so, continue to review evacuation-related procedures in light of changes in demographics and conditions. FEMA's specific findings on the exercise will be issued later this year, but the preliminary assessment indicates that the offsite emergency plans are adequate to protect public health and safety.

The Petitioners refer to the 1982 Sandia National Laboratory (SNL) Report, "Calculation of Reactor Accident Consequences" (CRAC-2 Report), and cite this report as a basis for concern that a terrorist attack could result in a massive release of radioactive materials. The reactor siting studies in the CRAC-2 Report were performed as part of research on the sensitivity of various plant siting parameters. The studies used generic postulated releases of radioactivity from a spectrum of severe (core melt) accidents, independent of the probabilities of the event occurring or the impact of mitigation mechanisms. The studies were never intended to be realistic assessments of accident consequences. The estimated deaths and injuries resulted from assuming the most adverse condition for each parameter in the analytical code. In the cited studies, the number of resulting deaths and injuries also reflected the assumption that no protective actions were taken for the first 24 hours. The studies did not, and were never intended to, reflect reality or serve as a basis for emergency planning. The CRAC-2 Report analyses used more simplistic models than current technologies. The two basic conclusions from the SNL siting studies were that the mean estimated number of health effects from the assumed releases for all reactor sites varied by up to more than 4 orders of magnitude and that the financial costs of the releases were dominated by clean-up costs and

replacement power costs. The SNL studies provided a useful measure to compare sites, not to analyze plant-specific accident consequences.

Regarding the Petitioners' assertion that the emergency plans do not contemplate multiple attacks on the infrastructure (i.e., roads, bridges, transportation, communications, etc.), the NRC finds that the existing emergency response plans allow considerable flexibility to respond to a wide variety of adverse conditions, including the results of a terrorist attack. As previously discussed in this Director's Decision, the NRC considers that commercial nuclear power plants have sufficient security measures in place to defend against a broad spectrum of potential terrorist threats, thereby precluding the release of radioactive material to the environment. If a terrorist attack inflicted damage on a nuclear plant, the redundant design features inherent in the plant, and the high level of training accorded the plant staff, would likely result in actions being taken by the plant staff to prevent or minimize the release of radioactive material. In the unlikely event of a significant release of radioactive material, for whatever reason, the emergency response plans provide for protective actions for the surrounding population. While the emergency response plans provide alternative actions in the event of some failures of the local infrastructure, there are limits to the degree to which it is reasonable to assume that infrastructure components are unavailable. The responsibility to preclude the large scale and resource intensive effort that would be required for a successful terrorist attack on multiple targets, rests with agencies of the Federal government. The NRC considers the actions of various intelligence and law enforcement agencies, combined with the actions of the Department of Defense, to provide assurance that a successful large scale terrorist attack is unlikely. Additionally, the NRC advisories and the Orders issued since September 11, 2001, directed licensees to take specific actions to improve existing emergency response plans, including heightened coordination with local, State, and Federal authorities. In

summary, the NRC concludes that emergency preparedness plans and evacuation planning are routinely revised and updated, and are appropriate to use in response to a radiological emergency, including a release caused by a terrorist attack.

Information about Security Measures to Protect Against Terrorist Attacks

The Petitioners requested that the NRC require the licensee to provide information documenting the existing and readily attainable security measures which provide IP with protection against land, water, and airborne terrorist attacks. This information should provide sufficient basis for the NRC to determine that physical barriers, intrusion alarms, and other measures are in place or constructed and are sufficient to meet realistically expected threats.

Staff's Response

As previously discussed, the NRC and its licensees have taken a number of steps since September 11, 2001, to increase security at NRC-licensed facilities, including safeguards advisories. At IP, the licensee's security force was augmented by the New York State Police and the National Guard (including Hudson River patrols) and local law enforcement personnel.

The NRC issued Orders on February 25, 2002, to all commercial nuclear power plants to implement interim compensatory security measures for the current threat environment. Some of the requirements made mandatory by the Orders formalized the security measures that NRC licensees had taken in response to advisories issued by the NRC in the aftermath of the September 11 terrorist attacks. The Orders also imposed additional security enhancements, which have emerged based on the NRC's assessment of the current threat environment and its ongoing security review. The requirements will remain in effect until the NRC determines that the level of threat has diminished, or that other security changes are needed. The specific actions are sensitive, but include increased patrols, augmented security forces and capabilities, additional security posts, installation of additional physical barriers, vehicle checks at greater stand-off distances, enhanced coordination with law enforcement and military authorities and more restrictive site access controls for all personnel. The Orders also require additional security measures pertaining to the owner-controlled land outside of the plants' protected areas. Currently, the New York State Naval Militia provides security measures to detect and deter watercraft access from entering the exclusion area around the IP plants.

In its report on security, the State of New York Office of Public Security (OPS) provided recommendations to enhance security at IP. Many of the measures suggested have been implemented by the licensee and others are currently under advisement. The measures are

recommendations by OPS to further enhance security and are not requirements in current NRC regulations. In response to the NRC Orders of February 25, 2002, the licensee provided information, that taken in conjunction with other sources of security information, resulted in the NRC finding the licensee's security posture to be appropriate under the current circumstances.

The Petitioners additionally seek specific details of security measures in place to respond to the potential for terrorist attacks. The NRC's policy is to withhold safeguards information from the public. Therefore, this request is denied.

Mandate Security Measures Sufficient to Protect the Facility

The Petitioners requested the NRC to mandate, at a minimum, the following security measures sufficient to protect the facility:

1. Obtainment of a permanent no-fly zone from the Federal Aviation Administration (FAA) in the air space within 10-nautical miles of the IP facility.
2. A defense and security system sufficient to protect and defend the no-fly zone.
3. A defense and security system sufficient to protect the entire facility, including the containment and spent fuel storage buildings, control room and electrical equipment.

Staff's Response

In the aftermath of September 11, 2001, the Federal government took a number of steps to improve aviation security and minimize the threat of terrorists using airplanes to damage facilities critical to our nation's infrastructure. The Commission views that the efforts associated with protecting our nation from terrorist attacks by air should be directed toward enhancing security at airports and on airplanes. Thus, the Commission endorses the prompt response by Congress to strengthen aviation security under the Aviation and Transportation Security Act of 2001, because this legislation provides for improved protection against air attacks on all industrial facilities, both nuclear and non-nuclear. The NRC further supports the

steps taken by the FAA to improve aircraft security, including enhanced passenger and baggage screening, strengthening of cockpit doors, and the Air Marshal program. The U.S. intelligence community and various Federal law enforcement agencies have also increased efforts to identify potential terrorists and prevent potential attacks before they occur. For example, the FAA and Department of Defense have acted more than once to protect airspace above nuclear power plants from what were thought to be credible threats against certain specific sites. These potential threats were later judged to be non-credible.

The NRC is also reviewing measures to bolster defenses and to establish new antiterrorism strategies in a thorough and systematic manner. The NRC is taking a realistic and prudent approach toward assessing the magnitude of the potential threat and the strength of licensee defenses.

NRC licensees must defend nuclear power plants against the DBT. September 11 showed that the NRC and its licensees must reevaluate the scope of potential assaults of all types. However, there are limits to what can be expected from a private guard force, even assisted by local law enforcement. Even if it is determined that nuclear power plants should be defended against aircraft attack, the NRC cannot expect licensees to acquire and operate antiaircraft weaponry. Protection against this type of threat may be provided by other means within the Federal government.

In summary, the Petitioner's request is denied because the NRC considers that the collective measures taken since September 11, 2001, provide adequate protection of public health and safety.

Dry-Cask Spent Fuel Storage System

The Petitioners requested that the NRC order the licensee to immediately convert the current spent fuel storage from water-cooled SFPs to a dry-cask storage system in a bunkered

structure. As the basis for the request, the Petitioners state that this action would reduce the long-term risk of potential exothermic oxidation in the existing fuel storage facility. The Petitioners state that the NRC has never established that the spent fuel storage facility at IP is secure against foreseeable attacks nor can the NRC be certain that the spent fuel storage facility is sufficiently sound to preclude the possibility of a spent fuel fire in the event of an airborne, land, or water-based assault. The Petitioners' concerns were based, in part, on information in an NRC report, "Final Technical Study of Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," dated October 2000, and on the Petitioners' evaluation of the consequences of a terrorist attack on the spent fuel pool buildings. In their December 20, 2001, supplement, the Petitioners state that the NRC has not performed an environmental impact statement or probabilistic risk analysis assuming all modes of water loss from the SFPs, including terrorist attack, and the Petitioners further discuss the probability and consequences of exothermic oxidation of the spent fuel cladding.

Staff's Response

The NRC staff presently concludes that spent fuel can be safely stored at the IP reactor site in the current system of SFPs and therefore, the Petitioners' requests are denied. Although the spent fuel storage buildings at IP are not as hardened as the reactor containment structures, the SFPs themselves are robust, and relatively small structures, that are partially below ground level. The spent fuel is stored in racks resting on the floor of the pools and is covered by more than 20 feet of water. The pools are designed to prevent a rapid loss of water with the structure intact, and the pool water level and cooling system are monitored and alarmed in the control rooms. Thus, the response time for events involving the SFP is significantly longer than for other event scenarios. It is also easier to add water to the SFP from various sources because it is an open pool. The robust design and small size of the

pools minimize the likelihood that a terrorist attack would cause damage of a magnitude sufficient to result in an offsite release of radioactive material. Further, offsite resources can be brought onsite to assist the response to an event.

When the NRC staff completes its reevaluation of the physical security requirements, the NRC will be able to judge whether modifications to the SFP structures and enclosures are warranted and whether additional safeguards measures should be established. If so, the NRC will act accordingly. In the meantime, the NRC has issued Orders to all nuclear power plants requiring certain interim compensatory measures to augment security and strengthen mitigation

strategies. The SFPs are within the protected area of the facility and therefore protected from certain external threats under the security provisions identified in the PSPs.

During the NRC review of the transfer of the licenses for IP1 and 2, the licensee indicated that it was evaluating the possible construction of an independent spent fuel storage facility. In a public meeting on March 14, 2002, the licensee stated that it was expediting its engineering review for this facility.

III. Conclusion

As stated in its letter to the Petitioners on December 20, 2001, the NRC has, in effect, partially granted the Petitioners' request for an immediate security upgrade at IP2 and 3. On September 11, 2001, the NRC took action to enhance security at all nuclear facilities, including IP2 and 3. Immediately after the attacks, the NRC advised all nuclear power plants to go to the highest level of security, which they promptly did. These facilities have remained at a heightened security level since. The NRC continues to work with other Federal agencies and is monitoring relevant information it receives on security matters at nuclear facilities. The NRC is prepared to make immediate adjustments as necessary to ensure adequate protection of the public.

On February 25, 2002, the NRC issued Orders to IP2 and 3 and all other operating commercial nuclear power plants to implement interim compensatory security measures for the

high-level threat environment. Some of the requirements formalized a series of security measures that NRC licensees had taken in response to advisories issued by the NRC, and others are security enhancements that have emerged from the Commission's ongoing comprehensive security review. The Commission issued the Orders, which incorporated the threat advisories and added additional requirements, to formalize the security enhancements at commercial nuclear power plants. Because the threat environment had persisted longer than expected, it is appropriate to maintain these security measures within the established regulatory framework. The details of these security requirements are sensitive and will not be provided to the public. Some of the specific measures implemented by the licensees in response to the advisories and interim compensatory measures included increased patrols, augmented security forces and capabilities, additional security posts, installation of additional physical barriers, vehicle checks at greater stand-off distances, enhanced coordination with law enforcement and military authorities and more restrictive site access controls for all personnel. Therefore, the Petitioners' request that the licensee conduct a full review of the facility's vulnerabilities, security measures, and evacuation plans has been, in effect, partially granted. Regarding the Petitioners' request for specific information about the security measures, the NRC believes that it is inappropriate to discuss perceived vulnerabilities and current or planned security measures in the public domain. Thus, this request is denied.

The NRC in its February 25, 2002, Orders also directed licensees to evaluate and address potential vulnerabilities to maintain or restore cooling to the core, containment, and SFP and to develop specific guidance and strategies to respond to an event that damages large areas of the plant due to explosions or fires. These strategies are intended to help licensees to identify and utilize any remaining onsite or offsite equipment and capabilities. If

NRC's ongoing security review recommends any other security measures, the NRC will take appropriate action.

The NRC denies the Petitioners' request to mandate certain security measures, as specified by the Petitioners, for the protection of the facility, such as systems to defend a no-fly zone. As part of its ongoing comprehensive security review, the NRC is examining the threat environment in coordination with the new Office of Homeland Security, the FBI, FEMA, the FAA, the military, the intelligence community, and the Department of Energy, among others. The NRC will take appropriate action based on the results of this review. The NRC considers that the current security requirements, along with the enhancements in the February 25 Orders, provide reasonable assurance of the protection of the facility.

The NRC finds that the existing emergency response plans are flexible enough to respond to a wide variety of adverse conditions, including a terrorist attack. The NRC advisories and the Orders issued since September 11, 2001, directed licensees to take specific actions deemed appropriate to ensure continued improvements to existing emergency response plans. The Petitioners' concern that the emergency plans do not contemplate multiple attacks on the infrastructure is alleviated by the fact that the emergency plans are intended to be broad and flexible enough to respond to a wide spectrum of events. Thus, the Petitioners' request that the onsite and offsite emergency plans be revised to account for possible terrorist attacks has been, in part, granted.

The NRC finds that the current spent fuel storage system and the security provisions at IP adequately protect the spent fuel. However, the licensee has stated its intention to install an Independent Spent Fuel Storage Installation. The Petitioners' request to order the installation is denied.

As provided in 10 CFR 2.206(c), a copy of this Director's Decision will be filed with the Secretary of the Commission for the Commission to review. As provided for by this regulation, the Decision will constitute the final action of the Commission 25 days after the date of the

Decision unless the Commission, on its own motion, institutes a review of the Decision within that time.

Dated at Rockville, Maryland, this 18th day of November 2002.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

RAS 5101

December 18, 2002

DOCKETED 12/18/02
SERVED 12/18/02

Mr. Alex Matthiessen
Executive Director
Riverkeeper, Inc.
25 Wing & Wing
Garrison, NY 10524

Re: Director's Decision (DD-02-06)
Indian Point Nuclear Generating Unit
Nos. 1, 2 and 3
50-003, 50-247 and 50-286 (2.206)

Dear Mr. Matthiessen:

This is to inform you that the time provided by NRC regulation within which the Commission may act to review the Director's Decision (DD-02-06) has expired. The Commission has declined any review. Accordingly, the decision became final agency action on December 13, 2002.

Sincerely,

/RA/

Annette L. Vietti-Cook

cc: Service List

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket Nos. 50-003/247/286 (2.206)
	)	
	)	
(Indian Point Nuclear Generating,	)	
Units 1, 2 and 3)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing LETTER FROM ANNETTE L. VIETTI-COOK TO ALEX MATTHIESSEN RE DD-02-06 have been served upon the following persons by U.S. mail, first class, or through NRC internal distribution.

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[Original signed by Adria T. Byrdsong]

Office of the Secretary of the Commission

Dated at Rockville, Maryland,
this 18th day of December 2002

Before the
UNITED STATES NUCLEAR REGULATORY COMMISSION
Washington, D.C. 20555

In the Matter of:	:	TO: EXECUTIVE DIRECTOR FOR OPERATIONS
	:	
ENTERGY CORPORATION	:	
(Indian Point Nuclear Power Station,	:	Docket No. ____ __
Units No. 2 and 3; Facility Operating	:	
Licenses DPR-26 and DPR-64)	:	
----------		November 8, 2001

RIVERKEEPER, INC., et al,
Petitioners

SECTION 2.206 REQUEST FOR EMERGENCY SHUTDOWN

OF INDIAN POINT UNITS 2 AND 3

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Before the
UNITED STATES NUCLEAR REGULATORY COMMISSION
Washington, D.C. 20555

In the Matter of:	:	TO: EXECUTIVE DIRECTOR FOR OPERATIONS
ENTERGY CORPORATION (Indian Point Nuclear Power Station, Units No. 2 and 3; Facility Operating Licenses DPR-26 and DPR-64)	:	Docket No. _____
		November 8, 2001

RIVERKEEPER, INC., et al,
Petitioners

SECTION 2.206 REQUEST FOR EMERGENCY SHUTDOWN OF INDIAN POINT UNITS 2 AND 3

I. Request for Action

Riverkeeper, Inc. and the individual and organizational petitioners identified on the attached page (collectively, "Petitioners") hereby respectfully request, pursuant to 10 CFR §§ 2.206 and 2.202, that the United States Nuclear Regulatory Commission take the following immediate actions:

1. Order the Indian Point licensee to suspend operations, revoke the operating license, or adopt other measures resulting in a temporary shutdown of Indian Point Unit 2 and Unit 3, as per 10 CFR § 2.202, and order the licensee to conduct a full review of the facility's vulnerabilities, security measures and evacuation plans.

2. Require the licensee to provide information, as contemplated by 10 CFR § 2.204(a), documenting the existing and readily attainable security measures which provide the Indian Point facility with protection against land, water, and airborne terrorist attacks. Such information should provide, at a minimum, sufficient basis for the Commission to determine that physical barriers, intrusion alarms, and other measures are in place or may be easily constructed, and are sufficient to meet realistically expected threats.

3. Immediately modify the licensee's operating license for Units 2 and 3 to mandate, at minimum, the following security measures sufficient to protect the facility as required by 10 CFR § 73.55:

a. obtainment of a permanent no-fly zone from the Federal Aviation Administration in the air space within 10 nautical miles of the Indian Point facility;

b. a defense and security system sufficient to protect and defend the no-fly zone;

c. a defense and security system sufficient to protect the entire facility, including the containment and spent fuel storage buildings, control room and electricity equipment, from a land or water based terrorist attack. The security review described above should contemplate retaining these measures on a permanent basis, and/or discuss reasonable alternatives of equal efficacy.

4. Order the revision of licensee's Emergency Response Plan and Westchester County's Radiological Emergency Response Plan in order to account and prepare for possible terrorist attacks. These reviews must contemplate not only realistic and catastrophic effects of a terrorist attack on the Indian Point facility, but a comprehensive response to multiple attacks in the region which may impair the efficient evacuation of the area. Examples of such attacks include destruction of the Tappan Zee Bridge, loss of power to passenger railroads, and other events which deny use of necessary infrastructure.

5. If, after conducting a full review of the facility's vulnerabilities, security measures and evacuation plans, the NRC cannot sufficiently ensure the security of the Indian Point facility against terrorist threats, the Commission should take prompt action to permanently retire the facility.

6. Separate and apart from the above, the Commission must order the Indian Point licensee to undertake the immediate conversion of the current spent fuel storage technology from a water cooled system to a dry cask system in a bunkered structure in order to reduce the long-term risk associated with potential exothermic oxidation within the existing spent fuel storage facility.

As explained more fully below, the Indian Point facilities' containment structures, reactor vessels, spent fuel storage areas, control rooms, and electrical switching equipment are all vulnerable to terrorist attack. Indian Point, located in Westchester County, New York, is not currently equipped to defend itself, nor the 20 million people who reside and work within a 50 mile radius of the plant, against an attack of the scale, sophistication, and coordination demonstrated on September 11, 2001. A successful attack on these structures would have a catastrophic effect on the region's human population, environment, and economy. Based on this threat, Petitioners are requesting, among other things, that the United States Nuclear Regulatory Commission suspend the operating licenses for all units until such time as the licensee can demonstrate that the facility is protected against plausible attack scenarios.

II. The Interests of Petitioners

Petitioner Riverkeeper, Inc. is a not-for-profit organization whose mission is to protect the environmental, recreational and commercial integrity of the Hudson River, and to safeguard New York City's and Westchester County's drinking water supply. Petitioners together and independently state that they are personally affected and aggrieved by the continued operation of Indian Point without the specific security measures identified in this request.

Section 2.206(a) of Title 10, CFR, states that "[a]ny person may file a request to institute a proceeding pursuant to § 2.202 to modify, suspend, or revoke a license, or for any other action as may be proper." Riverkeeper, Inc. hereby submits this petition identifying the threat of a terrorist attack on the Indian Point facility as a new, site-specific, hazardous condition that is larger and more dangerous than previously considered in the licensing and the design basis threat of Indian Point Units 2 and 3.

III. Critical New Information Constituting the Basis for This Request.

A. The Indian Point Facility is a Plausible Target of Future Terrorist Actions.

The United States is currently facing a heightened state of security related to the recent terrorist attacks against infrastructure targets in New York City and Washington D.C. As political, judicial, and military operations against suspected terrorist organizations continue, civilian and military establishments within the United States remain plausible targets of future terrorist attack. New York City remains a primary terrorist target, as evidenced by the growing number of Anthrax cases in that city and its environs. As New York City is a terrorist target, so too are nearby industrial facilities that, if compromised, could cause devastation to the populace, environment, and economy. No other facility in the country, let alone in New York, poses as great a risk to as great a number of people as the Indian Point nuclear power plant. Among the factors making the Indian Point facility a plausible target for a terrorist attack is the facility's proximity to:

- A population density of approximately 20 million people within 50 miles of the facility;[1]

[1] According to year 2000 U.S. Census Bureau data, 19,086,634 people live within the twenty-six counties that are within 50 miles of Indian Point, *available at* http://www.census.gov/index.html.

000055

- Major financial centers in New York City that are essential to the functioning of the U.S. economy, (the greater New York City metropolitan area has the 14th largest economy in the world and the greatest economy of any metropolitan region in the nation);[2]
- The Croton, West Branch and Kensico reservoirs which supply and store nearly all of Westchester County's and most of New York City's drinking water; and
- Major air, sea, rail, and highway transportation systems that are vital to the regional and national economy.

B. Actual Threats Against Nuclear Power Plants Have Been Documented.

The imminent threat of a terrorist attack on a nuclear power plant is evidenced in news and in government statements. On November 3, 2001, Daniel Rubin reported in his article, *Nuclear Terrorism Threat Growing*: "[t]he vulnerability of power plants moved to center stage after last Sunday, when Canadian authorities monitored a phone call from an alleged al-Qaida member to Afghanistan. Two targets, he said, would be attacked this week 'down south,' including an unnamed nuclear facility."[3]

Shortly following the attacks on September 11, 2001 the Three Mile Island nuclear power plant received a "credible threat" on October 17, prompting officials to shut down two nearby airports and dispatch military aircraft to protect the facility.[4]

On November 1, 2001, Mohamed ElBaradei, Director General of the International Atomic Energy Agency (IAEA) warned that there is "the potential of terrorists targeting nuclear facilities."[5] Mr. ElBaradei also stated that the "safety and security of nuclear material is a legitimate concern of all States" and that "[t]he willingness of terrorists to commit suicide to

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[2] *See* The United States Conference of Mayors, *If U.S. City/County Metro Economies Were Nations, at* http://www.usmayors.org/citiesdrivetheeconomy/chart2_decade.pdf (last visited Nov. 7, 2001) (rankings based on gross domestic and metropolitan product 2000). Attached as Exhibit 1.

[3] Daniel Rubin, *Nuclear Terrorism Threat Growing*, KNIGHT RIDDER FOREIGN SERVICE (November 3, 2001) *available at* www.nci.org/01/11f/3-5.htm. Attached as Exhibit 2.

[4] CNN, *Threat at Three Mile Island Closes Airports*, October 18, 2001, *available at* www.cnn.com/2001/us/10/18/gen.three.mile.island/index.htm. Attached as Exhibit 3.

[5] International Atomic Energy Agency, *Calculating the New Global Nuclear Terrorism Threat* (November 1, 2001) *available at* www.iaea.org/worldatom/Press/P_release/2001/nt_Pressrelease.shtml. Attached as Exhibit 4.

achieve their evil makes the nuclear terrorism threat far more likely than it was before September 11."[6]

On November 1, 2001, the Washington Post reported: "Nancy Savage, an FBI agent in Eugene, Ore., who is president of the FBI Agents Association, said the biggest concerns for investigators include airports, power plants and other key infrastructure points."[7]

On July 4, 2001, the New York Times reported that an Algerian man, Ahmed Ressam, convicted of attempting to carry out a terrorist attack in Los Angeles, testified that he was trained in an Afghanistan camp run by Osama bin Laden and received training in how to blow up "the infrastructure of a country."[8] Ressam described how he was among 50 to 100 men at the camp who were being trained in "urban warfare."[9] Ressam stated that power plants were targets as they were labeled "enemies' installations."[10]

On October 21, 2001, the Sunday London Times reported that the FBI is studying a report that the four terrorists who seized Flight 93, which crashed near Pittsburgh, may have been targeting a nuclear power plant.[11] Most recently, the Federal Aviation Administration established a no-fly zone around nuclear power plants.[12] On October 30, 2001, the Washington Post reported on an interview with a jailed disciple of Osama bin Laden who said there are "more important places, like atomic plants and reactors" that may have been more appropriate targets than the World Trade Center.[13]

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[6] *Id.*

[7] Eric Pianin and Dan Eggen, *Preparations Stepped Up For Possible New Attacks*, WASHINGTON POST (November 1, 2001), *available at* www.washingtonpost.com/wp-dyn/articles/A20995-2001Oct.31.html. Attached as Exhibit 5.

[8] Lara Mansnerus and Judith Miller, *Terrorist Details His Training in Afghanistan*, N.Y. TIMES (July 4, 2001), *available at* www.nci.org/01/07/04-nyt-terrorist_afg.htm. Attached as Exhibit 6.

[9] *Id.*

[10] *Id.*

[11] Nicholas Rufford, David Leppard and Paul Eddy, *Nuclear Mystery: Crashed Plane's Target May Have Been Reactor*, SUNDAY TIMES (London, UK) (October 21, 2001), *available at* www.sunday-times.co.uk/news/pages/sti/2001/10/21/stiusausa02018.html. Attached as Exhibit 7.

[12] CNN, *FAA Restricts Flights Near World Series, Nuclear Plants* (October 30, 2001), *available at* http://edition.cnn.com/2001/us/10/30/gen.attack.on.terror. Attached as Exhibit 8.

[13] William Branigin, *In Afghan Jail, a Terrorist Who Won't Surrender*, WASHINGTON POST (October 30, 2001), *available at* www.washingtonpost.com/wp-dyn/articles/A8758-2001Oct29.html. Attached as Exhibit 9.

The NRC acknowledges the threat against nuclear power plants, as evidenced by the agency's move to shut down its website within weeks after the September 11 attacks. NRC's website is back up but with limited access to sensitive information, raising the question of whether terrorists already have downloaded and made use of information that has since been removed from the website. The United States remains on high alert in anticipation of additional terrorist attacks. As this petition was being written, the U.S. Justice Department announced it had credible information that another round of terrorism is imminent. In response to this renewed threat, the Governor of the State of New York, George Pataki, dispatched additional National Guard reservists to the state's nuclear power facilities. However, the strained resources of the state and the National Guard cannot ensure sustained adequate protection from terrorist attacks.

C. Indian Point Is Currently Vulnerable to Catastrophic Terrorist Attack.

1. While Operational, Indian Point Is Unnecessarily Vulnerable.

a. Risks from Takeover of or Damage to Control Rooms

As long as the facility remains operational, the control rooms are a likely and vulnerable target for terrorist attack. Seizure or disability of the control rooms would dramatically increase the potential for the intentional or accidental destruction of the reactor core. A terrorist attack on the control rooms of the Indian Point facility would cause a loss of control of Units 2 and 3. Disruption of the off-site power supplied to Indian Point or an on-site cutoff of power directly to the control room could render the control room inoperable. Back-up diesel generators are also vulnerable and sometimes unreliable. An on-site fire affecting the control room could render the control inoperable or ineffective, as technicians would be forced to leave or die. A properly functioning control room and control staff is necessary to ensure safe operation of an active reactor core. Absent proper control of plant operations, the risk of a reactor core melt-down or spent fuel storage incident rises precipitously.

b. Breach of Operating Reactors Creates Greater Danger of Catastrophic Contamination.

As discussed below, the reactor containment walls were not designed to withstand the accidental or intentional crash of fuel laden jetliners. The incidents of September 11, 2001, have

introduced the likelihood that such a aircraft may be used against high-risk infrastructure facilities. The dangers posed by a breach of the containment domes of Indian Point Units 2 and 3 would be reduced by a cold shut down of these reactors. In particular, de-powering the reactors would reduce a potential release of high-risk radio-nucleides, thereby lowering long-term impacts such as childhood thyroid cancers of the type encountered in the wake of the Chernobyl accident.

c. Operational Facility Creates Multiple Vulnerable Points in Plant Security.

Because the reactors at Indian Point are operational, the licensee must extend full security measures to ensure protection of the control rooms and to guard against strikes that threaten the structural stability of the containment domes. While shutting down the reactors will not remove the need for such security, the reduction in threat would allow the licensee to focus its protective efforts on the more critical areas of the facility, especially the spent fuel storage area. The licensee would be able to shift some security personnel away from low risk areas, concentrating resources where they are most valuable and most likely to protect effectively against the deadliest attack.

d. Indian Point and NRC Personnel and Resources Confront Dual Challenges When Ensuring Security At Operational Facility.

Currently, employees at Indian Point must ensure both the safe and stable generation of power and create a heightened security environment. Simultaneously, NRC personnel are tasked with overseeing the ordinary operations of the plant while also ensuring that nuclear plants like Indian Point are protected against foreseeable threats. Resources of both the agency and the licensee have been stretched thin by this double-tasking.

On September 21, 2001, for example, the NRC announced that it was "working around the clock to ensure adequate protection of nuclear power plants and nuclear fuel facilities," and had directed the Staff to review NRC security regulations and procedures.[14] NRC also reports that it has advised all nuclear power plant licensees to maintain a state of the "highest level of

[14] NRC, Press Release No. 01-112, *NRC Reacts to Terrorist Attacks* (September 21, 2001). Attached as Exhibit 10.

security." In addition, the NRC conceded that nuclear power plants are not designed to withstand to withstand crashes by large aircraft.[15]

The Commission's efforts in this regard highlight the credibility of the threats faced by nuclear power plants, particularly Indian Point. The necessity for round the clock operations, even figuratively, demonstrates the levels of staff time and resource commitments necessary to meet currently foreseeable threats. Closing down reactor operations will reduce these expenditures, simplifying daily operations at Indian Point, thereby reducing the chances of an accident while allowing both the Commission and licensee to prioritize security measures.

e. Shutting Down Indian Point's Reactors Creates A More Secure Environment

Security of spent fuel has never been demonstrated at Indian Point. A provisional shut down of the plant is needed to allow the licensee and NRC to test critical security provisions for this facility. "Business as usual" operation of Indian Point provides no incentive for the plant's owner and NRC to remedy this long overlooked vulnerability.

2. Vulnerability of the Spent Fuel Storage Facility

Terrorist action against the spent fuel storage facility could result in a catastrophic failure of the containment system. NRC has never established that the Indian Point spent fuel storage facility is secure against foreseeable attacks. Likewise, the Commission cannot be certain that the structure of the storage facility is sufficiently sound to preclude the possibility of a spent fuel fire in the event of an airborne, land, or water based assault.

NRC has not properly evaluated the consequences of terrorist attack on the spent fuel storage area. In a study conducted by the NRC in October 2000, it stated that:

> "the risk analysis in this study did not evaluate the potential consequences of a sabotage event that could directly cause off-site fission product dispersion, for example, a vehicle bomb driven into or otherwise significantly damaging the SFP [Spent Fuel Pool], even after a zirconium fire was no longer possible."[16]

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[15] *Id.*

[16] NRC Report October, 2000 at 4-15. Attached as Exhibit 11.

A likely result of an aircraft crashing into a spent fuel storage facility, or of a truck bomb explosion similar to that which destroyed the Alfred E. Murrow Federal Building, would be a precipitous loss of cooling water in the spent fuel pools. During the course of normal operation, the presence of cooling water reduces heat produced by the decaying fuel rods and minimizes the potential for fire in the fuel cladding. In the absence of cooling water, adequate air circulation through the spent fuel storage racks is necessary to prevent such a fire. Partial dewatering of the storage pools will block this air flow, especially if the racks are damaged or obstructed by falling debris or the force of an explosion.

A reduction of cooling water in the spent fuel pools could lead to a catastrophic release of radiation. As the water in the fuel pool is reduced the remaining water will heat up and evaporate. This could expose the zirconium cladding which surround the spent fuel rods to oxygen and steam, resulting in an exothermic reaction that will lead to a spent fuel rod assembly fire. This event would release deadly amounts of radiological material and toxic fumes. The NRC October 2000 report stated:

> This reaction of zirconium and air, or zirconium and steam is exothermic (i.e., produces heat). The energy released from the reaction, combined with the fuel's decay energy, can cause the reaction to become self-sustaining and ignite the zirconium. The increase in heat from the oxidation reaction can also raise the temperature in adjacent fuel assemblies and propagate the oxidation reaction. The zirconium fire would result in a significant release of the spent fuel fission products which would be dispersed from the reactor site in the thermal plume from the zirconium fire. Consequence assessments have shown that a zirconium fire could have significant latent health effects and resulted (sic) in number of early fatalities.[17]

A Department of Energy report indicates that such a fire would release considerable amounts of cesium-137, an isotope that accounted for most of the offsite radiation exposure from the 1986 Chernobyl accident.[18] Another report, authored by NRC, concludes that, in the event of a pool fire approximately 100 percent of the pool's inventory of cesium would be released to the atmosphere.[19]

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[17] NRC Report October, 2000 at 3-1 (internal citation omitted).

[18] *See* US Department of Energy, Health and Environmental Consequences of the Chernobyl Nuclear Power Plant Accident, DOE/ER-0332 (Washington, DC: DOE. June 1987).

[19] *See* V L Sailor et al, Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82. NUREG/CR-4982 (Washington, DC: NRC, July 1987).

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The emission of radioactive particles from a spent fuel pool accident would lead to horrific consequences. The NRC study stated that human fatalities within the first year of such an event "can be as large as for a severe reactor accident even if fuel has decayed several years."[20] The radioactive fallout from this type of release could also make tens of thousands of acres of land uninhabitable.

An uncontrolled fuel rod fire in one pool could quickly cause fires in other pools where water loss is occurring. In the October 2000 report, the NRC stated that "[i]f the fuel handler fails to respond to the alarm or is unsuccessful in extinguishing the fire within the first 20 minutes, the staff assumes that the SSP cooling system will be significantly damaged and cannot be repaired."[21]

In addressing catastrophic events such as an earthquake, the report stated that the spent fuel pools "are also subject to unpredictable changes as a result of the severe seismic, cask drop and possibly other dynamic events which could rapidly drain the pool." A terrorist attack is one such dynamic event.

The spent fuel storage buildings at Indian Point are not capable of withstanding a terrorist attack. The roofs are "made partly out of pretty insubstantial metal, like sheet metal," according to the Nuclear Energy Institute.[22] This construction, coupled with relatively thin walls, is insufficient to protect against large vehicles or medium sized aircraft. The storage facilities are highly vulnerable to a ground-based attack of only several individuals or to a car and/or truck bomb. Compromise of the storage facility could pose an immediate health threat to workers and residents within close proximity of the Indian Point facility since radiation levels in the spent fuel storage facility can be five times higher than radiation levels in the containment area.[23]

The spent fuel storage area is highly vulnerable to an air attack and mitigation and control of damage from such an attack is highly improbable. An NRC report stated that an aircraft crashing into the spent fuel storage area could seriously affect the "structural integrity of the

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[20] *See* NRC Report October, 2000 at 3-34.

[21] *See* NRC Report October, 2000 at 3-16.

[22] *See* Vibeke Laroi, *Spent Nuclear Fuel Pools Seem Vulnerable To Attack*, REUTERS (Nov. 4, 2001).

[23] Based on calculations assuming that there are 15 times as many cores in the spent fuel storage area than in the containment area.

spent fuel pool or the availability of nearby support systems, such as power supplies, heat exchanges, or water makeup sources, and may also affect recovery actions."[24] The NRC study goes on to estimate that "1 of 2 aircrafts are large enough to penetrate a five foot thick reinforced concrete wall. The conditional probability that a large aircraft crashing penetrate a 5-foot-thick reinforced concrete wall is taken as 0.45."[25]

This probability is based on the occurrence of catastrophic damage to the spent fuel pool where "the pool is so damaged that it rapidly drains and cannot be refilled from either onsite or offsite resources."[26] Such an impact could cause a catastrophic event. The report estimates that a worse case scenario radiation release from a spent fuel rod fire will cause a 4.3 percent increase in early fatalities among those who are late to evacuate the one mile perimeter.[27] The individual risk of latent cancer fatalities from a worse case scenario release would be 8.42% higher.[28]

3. Design Basis Threat of the Indian Point Facility

The design basis threat did not consider the possibility of an intentional terrorist attack from the air or water, or a suicide attack from any front. The NRC has acknowledged that the Indian Point facility was not designed to withstand an attack by a fuel-laden, wide body jet. NRC spokesman, Neil Sheehan, stated "[w]e have not done the analysis, so we are not going to guarantee that a plane couldn't breach the containment."[29] The NRC news release of September 21, 2001 (No. 01-112), reads "the NRC did not specifically contemplate attacks by aircraft such as Boeing 757s or 767s and nuclear plants were not designed to withstand such crashes. Detailed engineering analysis of a large airline crash has not yet been performed."[30] Victor Dricks, NRC

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[24] NRC Report October, 2000 at 3-23.

[25] *Id.* at 3-23.

[26] *Id.* at 3-23.

[27] *Id.* at 3-29.

[28] *Id.*

[29] Roger Witherspoon, *Indian Point Chief: Plant Safe From Possible Attack*, JOURNAL NEWS (October 20, 2001), *available at* www.thejournalnews.com/newsroom/102001/20entergy.html. Attached as Exhibit 12.

[30] NRC Press Release, *NRC Reacts to Terrorist Attacks*, September 21, 2001.

spokesperson, stated that: "No one considered the possibility of suicide hijackers steering a large aircraft into a nuclear plant." [31]

In 1982, the U.S. Energy Department's Argonne National Laboratory performed a study that detailed the probable damage a jetliner could cause on the concrete containment walls protecting the reactors. While this study only addressed an accidental crash, it focused on the force of an impact into the primary containment wall and interior structure of a nuclear reactor. The report estimated that, if just one percent of a jetliner's fuel ignited after impact, an explosion inside the already damaged reactor building would occur generating a force equivalent to 1,000 pounds of dynamite.[32] The more fuel, the worse the explosion. The report stated that the ignition of fuel "could lead to a rather violent explosion and impose upon the primary containment relatively severe loads."[33]

The report added that U.S. nuclear regulators might have underestimated the potential damage from such explosions. The report also mentioned that "the breaching of some of the plant's concrete barriers may often be tantamount to a release of radioactivity." The report also stated that "[I]t appears that fire and explosion hazards have been treated with much less care than the direct aircraft impact. Therefore, the claim that these fire/explosion effects do not represent a threat to nuclear power plant facilities has not been clearly demonstrated."[34]

4. Defending the Indian Point Facility Against a Terrorist Attack

Security forces at nuclear power plants have repeatedly failed to repel mock terrorist attackers. In NRC recognized drills intended to test the ability of plants to defend against land based terrorist attacks, the nuclear industry has repeatedly failed to stop mock terrorist assaults from reaching the secure area of the plant and wreaking simulated damage that would, in a real situation, result in a core meltdown. The NRC reveals that 33 of the 68 facilities failed to repel

[31] Vibeke Laroi, *Nuclear Plant Scare Adds to U.S. Security Jitters*, REUTERS (October 18, 2001). Attached as Exhibit 13.

[32] U.S. Energy Department's Argonne National Laboratory Study, 1982.

[33] Id.

[34] Id.

small groups of mock intruders whose weapons, explosives, and tactics are severely limited.[35] In response to this problem, Paul Leventhal, founder of the Nuclear Control Institute,[36] stated:

> The security guards at half the nuclear power plants in the United States have failed to repel mock terrorist attacks against safety systems designed to prevent a reactor meltdown. These are so-called "force-on-force" exercises supervised by the Nuclear Regulatory Commission. The NRC refuses to take enforcement action in response to the failures, and is in the process of weakening the rules of the game in response to industry complaints. Sabotage of nuclear power plants may be the greatest domestic vulnerability in the United States today. This is the time to strengthen, not weaken, nuclear regulation.[37]

Despite Entergy's (the licensee of the Indian Point facility) assurances that it will be able to adequately protect against a terrorist attack, the reality of protection seems dubious.

5. Security And Safety Violations by Indian Point Licensee

It is clear from Entergy's history of violations that their claims of having sufficient security and the ability to protect the facility against a terrorist attack need to be called into question. Entergy has a demonstrably poor security record. As recently as August 2000, Entergy was sanctioned by the NRC for failure to maintain adequate physical protection of the Waterford 3 facility in Killona, Louisiana.[38] As a result of an October 1999 inspection by NRC staff, the NRC issued an order modifying Entergy's operating license in order to "assure that corrective actions are effectively implemented over the long term...and are necessary for [Entergy] to maintain compliance with 10 CFR 73.55[a]."[39] The NRC order explained that "[b]ased on the conduct of tabletop exercises, weaknesses were identified with the Licensee's capabilities to respond adequately to a design basis threat intrusion."[40]

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[35] Douglas Pasternak, *A Nuclear Nightmare*, U.S. WORLD & NEWS REPORT (September 17, 2001), *available at* www.nci.org/01/09/09-3.htm. Attached as Exhibit 14.

[36] The Nuclear Control Institute (NCI) is a non-partisan and non-profit Washington, D.C. based, independent research and advocacy center specializing in problems of nuclear proliferation.

[37] Paul L. Leventhal, Commencement Address to the Class of 2001 at Franklin & Marshall College (May 13, 2001). Attached as Exhibit 15.

[38] NRC Enforcement Action EA-00-093, dated August 4, 2000.

[39] *Id.* at 1 (regulation entitled "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage").

[40] *Id.* at 5.

In addition to this enforcement order, Entergy has been subject to several enforcement actions for inadequate physical protection at its other facilities. Such violations include knowingly providing false information to an NRC inspector concerning a failure to provide adequate escorts to visitors inside the vital area.[41] Entergy has also been cited for numerous other violations related to safety.[42]

Last year, Indian Point 2 became the first nuclear plant in the nation to be given a "red" designation, giving it the highest risk assessment in the nation. The NRC gave the plant its worst rating because of the operators failure to detect flaws in a steam generator tube before a radiation leak in February 2000.[43]

6. History of Emergency Preparedness Problems at the Indian Point Facility

The Indian Point facility has long history of safety problems related to the ability to respond to emergency situations. In Inside NRC, 2000, the Commission reported that a NRC Region One Automated Inspection Team (AIT) found emergency response problems at Indian Point. The report stated that the

> [e]mergency response data system, which links the site with the NRC's operations center, was in-operable for the first several hours of the event due to pre-existing equipment problems. The utility was slow to activate emergency facilities; beepers used to notify emergency response personnel; phone number contacts were outdated and confusion exists about who responded when, some responders entered Indian Point's site from a back gate to Indian point Three and their arrival and whereabouts were not noted.[44]

D. Impacts of a Terrorist Attack on Indian Point's Unit Two and Unit Three

A successful attack on either of Indian Point's reactors or spent fuel storage facilities would likely result in a massive release of radioactive materials into the surrounding towns and counties, quite possibly reaching into and contaminating New York City. Such a release would

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[41] Independent Action IA 96-052 against Richard M. Gracin, Security Shift Supervisor, dated December 19, 1996.

[42] History of NRC violations: NRC OE EA 96-274 $50,000 fine - maintenance violation; NRC OE EA 96-025 $50,000 fine - safety system violations; NRC OE EA 95-076 unauthorized people gaining unescorted access to secure areas ; NRC OE EA 94-161 unauthorized access to felons; NRC OE EA 94-105 $112,000 fine - safety violation penalty; NRC OE EA 93-071 $112,500 fine - violation of physical security and safety problems.

[43] *See* Shawn Cohen, *NRC Flags ConEd With Red Tag For Indian Point*, THE JOURNAL NEWS (November, 21, 2000). Attached as Exhibit 16.

[44] *See* NRC, *Inside NRC, 2000.*

cause hundreds of immediate fatalities close to the site and 100,000 or more latent cancer deaths farther downwind of the plant.[45] Further, a major release would probably contaminate the drinking water supply for New York City and Westchester County, devastate the area's ecology, and render portions of the New York metropolitan area uninhabitable.

1. Illness and Fatality Data

A study performed by NRC estimates that a terrorist attack on the Indian Point Unit 2 reactor that leads to a meltdown would cause "46,000 Peak Early Fatalities, 141,000 Peak Early Injuries, [and] 13,000 Peak Deaths from cancer."[46] A meltdown of the Indian Point Unit 3 reactors would cause "50,000 Peak Early Fatalities, 167,000 Peak Early Injuries, [and] 14,000 Peak Deaths from cancer."[47]

Loss of life and long-term illnesses will be exacerbated by the near-impossibility of evacuating the 22 million people who live within the 50 mile radius surrounding Indian Point. Following the 1979 accident at Three Mile Island, then Director of NRC's Office of State Programs testified that a similar accident at Indian Point would have had far more drastic consequences:

> Everybody says what a terrible situation we had at Three Mile Island, and I agree, but can you imagine what it would have been if it had been at Indian Point? It would have calamitous. You would have had dozens, hundred of people killed perhaps trying to get out of the place, because the roads are, you know, they're North-South roads basically and...there are narrow old bridges, one of the oldest bridges across the Hudson, the Bear Mountain Bridge, is a two-lane bridge...It's just a ridiculous place.[48]

2. Economic Loss Data

This same NRC study reveals that a terrorist attack on the Indian Point Unit 2 or 3 reactors that leads to a meltdown would cause $274 billion (1982 dollars) in property damage,[49]

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[45] Sandia Labs, NRC, Calculation of Reactor Accident Consequences (1982) ("CRAC-2 Report"). (We understand that the NRC has a more recent model or code for performing these estimates. We encourage the Commission to update these figures.)

[46] *Id.*

[47] *Id.*

[48] *At Indian Pt., A History of Nuclear Power, Problems And Controversy*, N.Y. TIMES (May 6, 1983). Attached as Exhibit 17.

[49] Sandia Labs. NRC. Calculation of Reactor Accident Consequences (1982).

and $314 billion (1982 dollars) in property damage respectively.[50] In terms of 2000 dollars, property damage from a Unit 2 meltdown would be estimated conservatively at $500.5 billion, and property damage from a Unit 2 meltdown would be estimated conservatively at $573.5 billion -- figures based solely on inflation without factoring the substantial rise in metropolitan area real estate values.[51] Data from the New York State Office of Real Property Services show that property values in Westchester County, and NY state in general, have increased four-fold since 1982.[52] Compounding this economic disaster would be the tremendous loss of both personal and corporate equity, and the loss resulting from uninsured and unrecoverable defaults on mortgage loans resulting from property loss.

An economic loss of this magnitude for the City of New York would have devastating consequences on our nation's entire economy.

3. Environmental Consequences

The potential dispersal of radiological contaminants into the water, atmosphere, and on land, would cause extensive and irreversible environmental damage. The dispersal of radiological contaminants is dependent on their physical and chemical properties. Some particles would be suspended or dissolved in water, contaminating drinking water supplies. The consumption of these suspended particles would adversely affect the health of aquatic life. Some radioactive isotopes are known to bio-accumulate in the tissues and organs of wildlife, thereby leading to systemic contamination of the food chain and further injury to humans. As some of the radioactive particles fall out of suspension and settle, river and reservoir beds would become contaminated. Furthermore a radioactively contaminated Hudson River would lose its recreational and commercial value as it would be unnavigable, unswimmable, and unfishable.

Particles that remain airborne would be respirable by humans and wildlife causing latent carcinogenic, mutagenic and teratogenic effects. Particles that settle out of the air would contaminate plant life and lands, causing lasting damage to entire ecosystems.

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[50] *Id.*

[51] *See* The Inflation Calculator, available at http://www.westegg.com/inflation/ (based on increases in the Consumer Price Index) (last visited Nov. 7, 2001).

[52] *See* New York State Office of Real Property Services, *Exemptions From Real Property Taxation In New York State* (1982 & 1999 reports), *available at* http://www.orps.state.ny.us/.

E. Westchester County's Radiological Emergency Preparedness Plan (REPP)

Westchester County's Radiological Emergency Preparedness Plan (REPP) was last revised in May of 2000 and does not address the site-specific, hazardous conditions of a sabotage event or a terrorist attack at the Indian Point facility. Moreover, the REPP does not address the likelihood of a meltdown event, a spent fuel storage area release, or a spent fuel assembly fire.

The REPP is flawed because it is based on erroneous assumptions. Therefore, REPP is inadequate in providing protection to the public. The assumptions in the REPP preclude the occurrence of an intentional act of terrorism or sabotage, a meltdown event involving Units Two or Three, the radiological release from the spent fuel storage area, a spent fuel rod fire, or the possibility of an explosion at the Indian Point facility.[53] A particularly disturbing assumption is that the effect of an accident "would almost certainly be contained within the reactor containment building. Nonetheless, an accidental release of radioactive materials to the off-site environment remains a remote possibility."[54] In the event of an accident, the REPP only considers the potential release of radioactive iodine, xenon, and krypton gases.[55] The REPP, by omission, ignores the release of cesium, stontium-90, plutonium and other radiological and toxic contaminants that will be released from a meltdown scenario, compromise of the spent fuel pools, or a spent fuel assembly fire.[56]

The inadequacies of emergency response at the Indian Point facility were known and considered decades ago, however no action was taken to resolve response problems. Robert Ryan, previous Director of NRC's Office of State Programs, said in a sworn statement after Three Mile Island, "it is insane to have a three-unit reactor on the Hudson River in Westchester County, 40 miles from Times Square, 20 miles from the Bronx" and that the emergency response

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[53] Indian Point Radiological Emergency Preparedness Plan for Westchester County, Revision May, 2000, at I-3.

[54] *Id.* at I-3, I-4.

[55] *Id.* at I-4.

[56] *Id.*

plan for serious accidents at the facility was a "nightmare."[57] Given the imminent threat of a terrorist attack on the Indian Point facility Mr. Ryan's statement is doubly true today.

Based on the inherent inadequacies of the REPP and its preclusion of the new site-specific, hazardous condition posed by a terrorist threat, the REPP needs to be revoked. In the interim, the Indian Point facility should be temporarily shut down until a realistic REPP can be developed, implemented, and tested.

F. Economic Impact of Actions Requested

The temporary shutdown of the Indian Point facility during the fall, winter, and spring will not have a significant impact on the supply and cost of electricity to consumers. The Indian Point reactors generate only five percent of the electricity in New York State. During the fall, winter, and spring months, there is a surplus of generating capacity in New York State. The potential costs increase to consumers from a temporary shutdown of Indian Point will be approximately 1/10 of one cent per kilowatt-hour. This is a small price to pay in comparison to the potential loss of life, environmental damage and economic loss that could result from terrorist attack at the Indian Point facility.

IV. The NRC Has Broad Discretionary Powers to Order and Implement Petitioner's Requests.

Pursuant to 10 CFR § 2.202(a) the Commission has authority to "institute a proceeding to modify, suspend or revoke a license or to take such actions as may be proper." In upholding its duty to protect the public, environment, and property, the NRC has broad discretionary powers to grant Petitioners' requests.

Section 161(b) of the Atomic Energy Act empowers the Commission to "establish rule[s], regulation[s], or order[s]" to "protect health or to minimize danger to life or property."[58] The NRC's authority to protect the public

> ...cannot be read simply to permit the Commission to provide adequate protection; another section of the Act "requires" the Commission to do that much.

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[57] *At Indian Pt., A History of Nuclear Power, Problems And Controversy*, N.Y. TIMES (May 6, 1983). Attached as Exhibit 17.

[58] 42 U.S.C. § 2201(b), (i).

> We therefore must view section 161 as a grant of authority to the Commission to provide a measure of safety above and beyond what is "adequate." The exercise of this authority is entirely discretionary. If the Commission wishes to do so, it may order power plants already satisfying the standard of adequate protection to take additional safety precautions.[59]

In addition, the Code of Federal Regulations, in Title 10, Sections 2.200, 2.204, 2.206, and 73.55, as well as other authority, authorizes the NRC to take the specific actions requested herein.

V. The Actions Requested Are Necessary and Appropriate to Protect the Safety of the Twenty Million People Living in the Vicinity of Indian Point.

Petitioners have properly "set forth the facts that constitute the basis for [this] request" pursuant to 10 CFR § 2.206, and have properly identified "the potentially hazardous conditions" as required by 10 CFR § 2.202(a)(1), specifically the threat of a terrorist attack on the Indian Point facility. The NRC should take immediate action in response to Petitioners' request.

The threat of terrorism on a scale of the September 11th attacks, and including assaults from the air or water, has not been previously considered in the licensing and/or design basis threat of the Indian Point facility, therefore, this request is proper and demands NRC's immediate attention and action. The Atomic Energy Act "commands the NRC to ensure that any use or production of nuclear materials 'provide[s] adequate protection to the health or safety of the public.'"[60] As of September 11, 2001, this duty has taken on a new dimension: the protection of the public from threat of a major radiological release resulting from a terrorist attack. Given that NRC's "paramount responsibility [is] protection of the public health and safety and the environment,"[61] the NRC should immediately order the actions requested herein and more fully articulated below.

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[59] *Union of Concerned Scientists v. NRC*, 824 F.2d 108, 110 (D.C. Cir. 1987).

[60] *Shoreham-Wading River Cent. Sch. Dist. v. U.S. NRC*, 931 F.2d 102, 106 (D.C. Cir. 1991) (*citing* 42 U.S.C. § 2232(a)).

[61] Conference Report at 47, 1982 U.S. Code Cong. & Admin.News at 3617.

A. The NRC Should Order An Immediate, Temporary Suspension of Operations at Indian Point Units Two and Three, and Conduct a Full Review of The Facilities' Vulnerabilities, Security Measures and Evacuation Plans.

Title 10, Section 2.202 of the Code of Federal Regulations authorizes the NRC to "modify, suspend, or revoke a license, or [take] any other action as may be proper."[62] The NRC should immediately order the Indian Point licensee to suspend operations, revoke the operating license, or adopt other measures resulting in a temporary shutdown of Indian Point Unit Two and Unit Three, as per 10 CFR § 2.202, and to conduct a full review of the facility's vulnerabilities, security measures and evacuation plans for the following reasons:

First, as explained above, as an operating facility, Indian Point is unnecessarily vulnerable to risks from takeover of or damage to control rooms. As long as the facility remains operational, the control rooms are a likely and vulnerable target for terrorist attack. Seizure or disability of the control rooms would dramatically increase the potential for the intentional or accidental destruction of the reactor core.

Second, because the reactor containment walls were not designed to withstand the accidental or intentional crash of fuel Laden jetliners, a breach of operating reactors creates a significantly greater danger of catastrophic contamination. The danger of a potential release of high-risk radio-nucleides would be reduced by a temporary de-powering and cold shut down of these reactors.

Third, the operating facility creates multiple vulnerable points in plant security. While shutting down the reactors will not remove the need for security, the reduction in threat would allow the licensee to focus its protective efforts on the more critical areas of the facility, especially the spent fuel storage area, thereby concentrating resources where they are most valuable and most likely to effectively protect against the deadliest attack.

Fourth, Indian Point and NRC personnel and resources confront dual challenges when ensuring security at an operational facility. Currently, employees at Indian Point must ensure both the safe and stable generation of power and create a heightened security environment. Simultaneously, NRC personnel are tasked with overseeing the ordinary operations of the plant – which is the only nuclear plant in the nation with a D rating (multiple/repetitive degraded

[62] 10 C.F.R. § 2.202.

cornerstone) from the NRC -- while also ensuring protection against foreseeable threats. Resources of both the agency and the licensee are stretched thin by this double-tasking.

Finally, shutting down Indian Point's reactors creates a more secure environment. Spent fuel security has never been demonstrated at Indian Point. A provisional shut down of the plant is needed to allow the licensee and the NRC to test critical security provisions for this facility. "Business as usual" operation of Indian Point provides no incentive for the plant's owner and the NRC to remedy this long overlooked vulnerability.

For all of these reasons, an immediate, temporary shutdown of the operating reactors at Indian Point Units Two and Three is necessary and prudent.

B. The NRC Should Require the Licensee to Provide Information Documenting That Existing Security Measures Are Sufficient Against Plausible Threats of Terrorist Attacks.

Title 10, Section 2.204(a) of the Code of Federal Regulations authorizes the NRC to demand information from a licensee. The NRC should immediately demand that Entergy provide information documenting the existing and readily attainable security measures which provide the Indian Point facility with protection against land, water, and airborne terrorist attacks. Such information must provide, at a minimum, sufficient basis for the Commission to determine that physical barriers, intrusion alarms, and other measures are in place or may be easily constructed to meet plausible threats, for the following reasons:

Actual threats against nuclear power plants have been documented and the Indian Point facility is a plausible target of future terrorist actions. However, as explained above, the design-basis threat for Indian Point did not consider the possibility of an intentional terrorist attack from the air or water, or a suicide attack from any front. Security forces at nuclear power plants have repeatedly failed to repel mock terrorist attackers. Moreover, Indian Point has a long history of safety problems related to the ability to respond to emergency situations. Entergy, the new owner and licensee of Indian Point, has a demonstrably poor security record and it is clear from Entergy's history of violations that its ability to protect the facility against a terrorist attack is questionable at best.

Because a terrorist attack was not considered in the plant's design basis threat, because mock attacks at nuclear plants are rarely thwarted by security forces, and because this facility and its operator have notoriously poor security histories, the NRC should immediately demand all information necessary to determine whether Indian Point is, or can be secured against a land-, air- or water-based terrorist attack.

C. The NRC Should Immediately Modify The Operating License For Units Two And Three To Mandate, At Minimum, Security Measures Sufficient To Protect The Facility as Required by 10 CFR § 73.55.

As explained above, a successful attack on either of Indian Point's reactors or spent fuel storage facilities would likely result in a massive release of radioactive materials into the surrounding towns and counties, quite possibly reaching into and contaminating New York City. Such a release would cause hundreds of immediate fatalities close to the site and 100,000 or more latent cancer deaths farther downwind of the plant. Further, a major release would probably contaminate the drinking water supply for New York City and Westchester County, devastate the area's ecology, and render portions of the New York metropolitan area uninhabitable.

Faced with this catastrophic threat, the NRC should, at a minimum, take action to obtain the following security measures:

1. a permanent no-fly zone within 10 nautical miles of the Indian Point facility;
2. a defense and security system sufficient to protect and defend the no-fly zone; and
3. a defense and security system sufficient to protect the entire facility, including the containment and spent fuel storage buildings, control room and electricity equipment, from a land- or water-based terrorist attack.

These measures are necessary to comply with the requirements of 10 CFR § 73.55 regarding physical protection of licensed activities in nuclear power reactors against radiological sabotage.

D. The NRC Should Order the Revision of Entergy's Emergency Response Plan and Westchester County's Radiological Emergency Response Plan in Order to Account for Critical New Information and Prepare for Possible Terrorist Attacks.

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The NRC should order the revision of Entergy's Emergency Response Plan and Westchester County's Radiological Emergency Response Plan in order to account and prepare for possible terrorist attacks. These reviews must contemplate not only realistic and catastrophic effects of a terrorist attack on the Indian Point facility, but a comprehensive response to multiple attacks in the region which may impair the efficient evacuation of the area. Examples of such attacks include destruction or blockage of the Tappan Zee Bridge, loss of power to passenger railroads, and other events which deny use of necessary infrastructure.

Westchester County's Radiological Emergency Preparedness Plan (REPP) was last revised in May of 2000 and does not address the site-specific, hazardous conditions of a sabotage event or a terrorist attack at the Indian Point facility.[63] Moreover, the REPP does not address the likelihood of class 9 event, a spent fuel storage area release, or a spent fuel assembly fire. The REPP is flawed as it is based on erroneous assumptions, therefore, REPP must be considered inadequate in providing protection to the public. The assumptions in the REPP preclude the occurrence of an intentional act of terrorism or sabotage, a class nine event involving Units Two or Three, the radiological release from the spent fuel storage area, a spent fuel rod fire, or the possibility of an explosion at the Indian Point facility.[64]

The inadequacies of emergency response at the Indian Point facility were known and considered decades ago, however no action was taken to resolve response problems. Based on the inherent inadequacies of the REPP and its preclusion of the new site-specific, hazardous condition posed by a terrorist threat, the REPP must be revised.

[63] Indian Point Radiological Emergency Preparedness Plan for Westchester County, Revision May, 2000.

[64] Id. at I-3, I-4.

E. The NRC Should Order The Licensee To Undertake The Immediate Conversion of the Current Spent Fuel Storage Technology From A Water Cooled System To A Dry Cask System.

As explained above, terrorist action against the spent fuel storage facility could result in a catastrophic failure of the containment system. NRC has never established that the Indian Point spent fuel storage facility is secure against foreseeable attacks, nor can it be certain that the structure of the storage facility is sufficiently sound to preclude the possibility of a spent fuel fire in the event of an airborne, land, or water based assault. A likely result of an aircraft crashing into a spent fuel storage facility or of a truck bomb explosion would be a precipitous loss of cooling water in the spent fuel pools.

A reduction of cooling water in the spent fuel pools could lead to a catastrophic release of radiation. As the water in the fuel pool is reduced the remaining water will heat up and evaporate. This could expose the zirconium cladding which surrounds the spent fuel rods to oxygen and steam, resulting in an exothermic reaction that will lead to a spent fuel rod assembly fire. This event would release deadly amounts of radiological material and toxic fumes leading to horrific consequences. Fallout from this type of release could make tens of thousands of acres of land uninhabitable. The spent fuel storage buildings at Indian Point are not capable of withstanding a terrorist attack. The spent fuel storage area is highly vulnerable to an air attack and mitigation and control of damage from such an attack is highly improbable.

However, an alternative is available that would greatly reduce, or even eliminate, the risk of a pool fire. Specifically, the fuel could be stored dry, in robust steel casks that are cooled by natural circulation of air, and each cask could be surrounded by an earth-and-gravel berm, with substantial spacing between the casks.[65] This storage arrangement would withstand a wide variety of determined acts of malice.[66] The design basis for this storage arrangement could include a requirement, among other things, that the impact of a large, fuel-Laden aircraft on the storage facility would not lead to a release of radioactive material from more than one cask.[67] A

[65] Telephone conversation with Ed Lyman, Nuclear Physicist at Nuclear Control Institute (Nov. 6, 2001).

[66] Id.

[67] Id.

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fuel storage facility constructed with such a design basis would not only be able to withstand or limit the consequences of a wide variety of acts of malice, but would also exhibit a very low probability of experiencing a substantial release of radioactive material due to events other than acts of malice.[68]

Consequently, pursuant to its power under 10 CFR § 2.202 to modify licenses or take other appropriate action, the NRC should order Entergy to undertake the immediate conversion of the current spent fuel storage technology from a water cooled system to a dry cask system.

CONCLUSION

The NRC is confronted with a new challenge: how to protect the nation's most densely populated area and the environment from the threat of a terrorist attack on the Indian Point facility. While this may be a challenging and daunting task, the NRC must react quickly and in a determined manner. The temporary shutdown of the Indian Point facility will significantly reduce the potential catastrophic consequences if it experiences a terrorist attack. As the potential harm resulting from such an attack is reduced, the Indian Point facility becomes less of an attractive target to terrorists. Therefore, the temporary shutdown and increased protection of the Indian Point facility is the most logical action to be taken to protect public health and minimize danger to life.

The NRC should order Entergy and local municipalities to review and update their emergency response plans in consideration and response to a terrorist threat since: (1) the threat of a large, highly coordinated terrorist attack has not been previously considered in the licensing or the design basis threat of the Indian Point facility; (2) it also has not been considered in the development of Indian Point's emergency response plan; and (3) it has not been considered in the Radiological Emergency Response Plans developed by local municipalities. If Entergy and/or any municipality determines that it is infeasible to develop a Emergency Response Plan or a Radiological Emergency Preparedness Plan to meet this new threat, then the NRC should order the shutdown of the Indian Point facility until the new site-specific, hazardous condition is abated.

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[68] Id.

Petitioners' requests are for reasonable and achievable measures that should be supported and implemented by the NRC. This is the only way NRC may uphold its congressional mandate to protect the lives, the environment and the property of the people of New York State.

Finally, in accordance with the Commission's petition guidance, Petitioners request a technical review meeting with the Petition Review Board (the "PRB"), including representatives of the Commission's Office of Nuclear Regulatory Research responsible for the Indian Point licenses at the earliest possible time and before any action is taken on this Petition.

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The Following Organizations and Individuals Join In the Foregoing Petition by Riverkeeper, Inc.

Nuclear Control Institute
STAR Foundation
Waterkeeper Alliance
Hudson River Sloop Clearwater

Eliot Engel, U.S. Congress (D)
Maurice Hinchey, U.S. Congress (D)
Jerrold Nadler, U.S. Congress (D)

Eric Schneiderman, NY State Senate (D)
Thomas Morahan, NY State Senator (R)
Suzi Oppenheimer, NY State Senate (D)
Richard Brodsky, NY State Assembly (D)
Samuel Colman, NY State Assembly (D)
Alexander Gromack, NY State Assembly (D)
Naomi Matusow, NY State Assembly (D)
Amy Paulin, NY State Assembly (D)
Ronald C. Tocci, NY State Assembly (D)

Stanley Michels, NY City Council (D)
Jim Gennaro, NY City Council (D)

Scott Vanderhoef, Rockland County Executive (R)
Tom Abinanti, Westchester County Board of Legislators (D)
George Latimer, Westchester County Board of Legislators (D)
Vincent Tamagna, Putnam County Board of Legislators (R)
Sam Oliverio, Putnam County Board of Legislators (D)
Harriet Cornell, Rockland County Board of Legislators (D)

Paul Feiner, Town Supervisor, Greenburgh (D)
Greenburgh Town Board
Charles Holbrook, Town Supervisor, Clarkstown (D)
John Dinin, Town Supervisor, Bedford (R)
Christopher P. St. Lawrence, Town Supervisor, Ramapo (D)

If U.S. City/County Metro Economies Were Nations

World Rankings on Gross Domestic and Metropolitan Product 2000 (U.S. Billions, Current)

THE UNITED STATES CONFERENCE OF MAYORS

Rank	Nation or Metro Area	GP 2000	Rank	Nation or Metro Area	GP 2000	Rank	Nation or Metro Area	GP 2000	Rank	Nation or Metro Area	GP 2000
1	United States	9,963.00	44	Finland	118.00	87	Cincinnati, OH-KY-IN	59.40	130	Syracuse, NY	30.10
2	Japan	4,614.00	45	Seattle-Bellevue-Everett, WA	115.00	88	Bergen-Passaic, NJ	59.30	131	Greenville-Spartanburg-Anderson, SC	29.90
3	Germany	1,873.00	46	Phoenix-Mesa, AZ	114.20	89	Indianapolis, IN	57.70	132	Jersey City, NJ	28.10
4	United Kingdom	1,410.00	47	Greece	110.90	90	Nigeria	54.90	133	Harrisburg-Lebanon-Carlisle, PA	27.10
5	France	1,286.00	48	Israel	108.00	91	Milwaukee-Waukesha, WI	54.80	134	Fresno, CA	26.30
6	China	1,104.00	49	SanFrancisco, CA	107.30	92	Las Vegas, NV-AZ	54.60	135	Omaha, NE-IA	26.20
7	Italy	1,074.00	50	Nassau-Suffolk, NY	106.80	93	San Antonio, TX	53.70	136	Tulsa, OK	25.70
8	Canada	699.00	51	San Diego, CA	104.60	94	Algeria	52.80	137	Albuquerque, NM	25.60
9	Brazil	665.00	52	Venezuela	102.90	95	New Zealand	52.10	138	Iraq	25.50
10	Mexico	578.00	53	Portugal	100.50	96	Norfolk-Virginia Beach-Newport News, VA-NC	51.70	139	Ventura, CA	24.50
11	Spain	557.00	54	Newark, NJ	96.30	97	Czech	50.80	140	Tucson, AZ	22.90
12	India	510.00	55	Baltimore, MD	96.20	98	Austin-San Marcos, TX	48.20	141	Akron, OH	21.90
13	Korea, South	480.00	56	Ireland	95.10	99	Buffalo-Niagara Falls, NY	47.80	142	Knoxville, TN	21.50
14	New York, NY	437.80	57	Singapore	93.70	100	Hungary	47.40	143	Toledo, OH	21.20
15	Australia	428.00	58	Oakland, CA	92.10	101	Fort Lauderdale, FL	46.70	144	Springfield, MA	20.90
16	Los Angeles-Long Beach, CA	363.70	59	Egypt	91.50	102	New Orleans, LA	46.50	145	Allentown-Bethlehem-Easton, PA	20.60
17	Netherlands	360.00	60	Denver, CO	91.10	103	Salt Lake City-Ogden, UT	46.40	146	Scranton-Wilkes-Barre-Hazelton, PA	20.60
18	Chicago, IL	332.80	61	Colombia	90.00	104	Greensboro-Winston-Salem-HighPoint, NC	46.30	147	Santa Rosa, CA	20.50
19	Taiwan	323.00	62	St.,Louis, MO-IL	89.60	105	Rochester, NY	45.70	148	Uruguay	20.49
20	Argentina	284.00	63	Malaysia	88.80	106	Richmond-Petersburg, VA	45.70	149	Baton Rouge, LA	20.40
21	Russia	247.00	64	SanJose, CA	85.40	107	Nashville, TN	45.20	150	Slovakia	20.20
22	Switzerland	241.30	65	Riverside-San Bernardino, CA	84.10	108	Raleigh-Durham-Chapel Hill, NC	44.30	151	Tunisia	19.96
23	Boston, MA	238.80	66	Tampa-StPetersb-Clrwater, FL	82.20	109	Jacksonville, FL	43.00	152	Dominican Republic	19.67
24	Belgium	227.00	67	Cleveland-Lorain-Elyria, OH	80.80	110	GrRapids-Muskegon-Holland, MI	42.30	153	Des Moines, IA	19.10
25	Sweden	224.10	68	Pittsburgh, PA	80.70	111	Memphis, TN-AR-MS	38.90	154	Ann Arbor, MI	19.10
26	Turkey	217.60	69	Philippines	78.00	112	Louisville, KY-IN	38.70	155	Columbia, SC	19.10
27	Washington, DC-MD-VA-WV	217.00	70	New Haven-BrPt-Stamford-Danbury-Waterbury, CT	76.80	113	Bangladesh	38.50	156	Guatemala	19.05
28	Austria	184.90	71	Chile	73.00	114	Kuwait	38.05	157	Tacoma, WA	19.00
29	Philadelphia, PA-NJ	182.40	72	Miami, FL	71.60	115	Albany-Schenectady-Troy, NY	37.80	158	Croatia (Hrvatska)	19.00
30	Houston, TX	177.50	73	Portland-Vancouver, OR-WA	71.50	116	Syria	35.53	159	Bakersfield, CA	18.90
31	Hong Kong	164.60	74	Iran	67.10	117	Morocco	34.80	160	Oman	18.82
32	Atlanta, GA	164.20	75	Puerto Rico	65.30	118	West Palm Beach-Boca Raton, FL	33.20	161	Fort Wayne, IN	18.60
33	Norway	164.00	76	Kansas City, MO-KS	64.80	119	Honolulu, HI	33.00	162	El Paso, TX	18.60
34	Poland	163.00	77	Hartford, CT	64.30	120	Monmouth-Ocean, NJ	33.00	163	Trenton, NJ	18.50
35	Dallas, TX	160.00	78	Middlesex-Somerset-Hunterdon, NJ	63.60	121	Romania	33.00	164	Slovenia	18.47
36	Denmark	158.00	79	Sacramento, CA	63.10	122	Providence-Warwick, RI	32.50	165	Little Rock-North Little Rock, AR	18.40
37	Detroit, MI	156.30	80	Fort Worth-Arlington, TX	63.00	123	Oklahoma City, OK	32.30	166	Madison, WI	18.40
38	Indonesia	147.60	81	Pakistan	62.70	124	Birmingham, AL	32.00	167	Lafayette, LA	18.20
39	Saudi Arabia	145.30	82	Peru	62.70	125	Ukraine	31.70	168	Kazakhstan	18.20
40	South Africa	132.30	83	Charlotte-Gastonia-Rock Hill, NC-SC	61.30	126	Wilmington-Newark, DE	31.40	169	Luxembourg	18.10
41	Orange County, CA	130.00	84	Columbus, OH	60.70	127	Dayton-Springfield, OH	31.20	170	Lexington, KY	17.80
42	Thailand	128.20	85	United Arab	60.70	128	Vietnam	30.60	171	Colorado Springs, CO	17.60
43	Minneapolis-St. Paul, MN-WI	121.30	86	Orlando, FL	59.50	129	Manchester-Nashua, NH	30.20	172	Wichita, KS	17.50

City/County Metro Areas Nations

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Nuclear terrorism threat growing

BY DANIEL RUBIN Knight Ridder Foreign Service

VIENNA, Austria -- After a month of dealing with anthrax fears, counterterrorism experts have their eyes on an even more ominous threat: a crude but effective nuclear attack.

U.S. intelligence and law enforcement officials say they have no evidence that Osama bin Laden's al-Qaida network or any other terrorist group has built, bought or stolen a nuclear weapon.

Terrorists, they say, are more likely to make a "dirty" bomb by mixing radioactive material with a conventional explosive or to attack a nuclear reactor with a truck bomb, plane or boat.

The vulnerability of power plants moved to center stage after last Sunday, when Canadian authorities monitored a phone call from an alleged al-Qaida member to Afghanistan. Two targets, he said, would be attacked this week "down south," including an unnamed nuclear facility.

"We now see nuclear terrorism to be a real possibility," said Mohamed El Baradei, director of the International Atomic Energy Agency, who Friday addressed a special U.N. session on nuclear terrorism here.

The agency is calling for international standards to protect radioactive materials from falling into the hands of extremists.

U.S. officials are concerned that two Pakistani nuclear scientists arrested by Pakistan might have given the al-Qaida network instructions how to

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build a dirty bomb with radioactive waste or enriched uranium, which associates of bin Laden reportedly have sought to acquire.

The most probable terrorist device could contain radioactive materials easily stolen from U.S. hospitals, research labs and industrial sites, said George Bunn, a professor at the Stanford University Center for International Security & Cooperation.

"If you explode a dirty bomb, you might not even kill anyone," he said, "but you would give everyone a real scare."

According to the Vienna-based IAEA, a watchdog agency that until Sept. 11 had focused on safety and keeping nuclear materials out of the hands of rogue nations, terrorists have never used a nuclear bomb, and bin Laden is not known to possess a nuclear capability.

Eighteen times since 1993, people have been found to be trafficking in highly enriched uranium or plutonium, the materials needed to make a nuclear bomb, the agency reported.

Less likely than dirty bombs, but far more devastating, would be attacks on nuclear reactors with truck bombs or planes, Bunn said.

In a recent Stanford survey of countries with peaceful nuclear programs, not one of the six respondents reported plans for dealing with truck bombs.

While nuclear power plants are built to withstand accidental crashes of small planes, a precision strike is another thing, Bunn said.

Another expert in nuclear terrorism, Matthew Bunn, an assistant director at Harvard's Kennedy School of Government, said he was most worried about "insiders" at nuclear facilities in Pakistan.

"It doesn't matter how many rings of men with guns you have around the place if an insider is working with terrorists," said Bunn, whose father is the Stanford professor.

He also voiced concerns about the loyalties of

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scientists in the former Soviet Union who live in 10 "nuclear cities" and earn the equivalent of $300 a month.

Asked which countries might unwillingly become unwitting sources for nuclear materials, he mentioned Yugoslavia, which he said has enough high-energy plutonium for a bomb, as well as Uzbekistan, Ukraine, Kazakstan and Latvia.

Many scientists at the conference have been warning about nuclear terrorism for years.

However, little money is spent on protecting against such threats, and the agency has no power to investigate the way nuclear materials are handled by the "nuclear weapons states" -- the United States, Russia, China, Great Britain and France.

And the other countries with nuclear capability -- Israel, India and Pakistan -- share little if any information.

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"You cannot impose any safeguards on them," said Morten Maerli, a researcher with the Norwegian Institute of International Affairs. "This is a major problem."

The agency has called for an international approach to what has been largely the job of individual countries -- and one shrouded in secrecy by those protecting military operations.

The agency, with a $100 million annual budget, says it needs to spend an additional $30 million to $50 million a year shoring up the safety of nuclear materials.

"This is a threat we basically know how to fix," Matthew Bunn said. "It's a matter of writing a check."

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Threat at Three Mile Island closes airports

HARRISBURG, Pennsylvania—The Three Mile Island nuclear power plant received a "credible threat" on Wednesday, prompting officials to shut down two nearby airports and dispatch military aircraft to protect the facility.

The plant was placed on a high state of alert as the FBI, state police and military planes scrambled to protect the facility.

Patrick von Keyserling, spokesman for the Dauphin County Emergency Management Agency, said the Nuclear Regulatory Commission got some kind of a threat against the nuclear facility and decided to restrict airspace in a 20-mile radius around the Harrisburg International Airport as a precaution. He did not have details on the threat.

Nuclear Regulatory Commission spokeswoman Diane Screnci declined to discuss the type of threat, or how the agency received it. The commission told Three Mile Island about the threat between 6 p.m. and 7 p.m., and the airports were shut down about 9 p.m., she said.

Ralph DeSantis, a spokesman for Three Mile Island, confirmed the plant's high state of alert, but also declined to discuss the type of threat or the additional security measures. He said Three Mile Island was the only nuclear power plant threatened. Three Mile Island is located just outside Harrisburg, which is 35 miles northwest of Lancaster.

Since the Sept. 11 terrorist attacks, the power plant had already taken additional security measures, but Wednesday's alert tightened security further, DeSantis and Screnci said.

The Associated Press contributed to this report.

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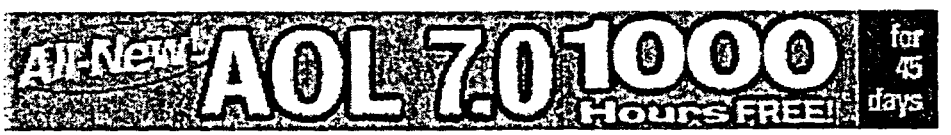

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The head of the International Atomic Energy Agency (IAEA) says that the ruthlessness of the 11 September attacks has alerted the world to the potential of nuclear terrorism - making it "far more likely" that terrorists could target nuclear facilities, nuclear material and radioactive sources worldwide.

Experts from around the world are meeting at the IAEA on 29 October to 2 November at an international symposium on nuclear safeguards, verification, and security. A special session on 2 November focuses on the issue of combating nuclear terrorism.

"The willingness of terrorists to sacrifice their lives to achieve their evil aims creates a new dimension in the fight against terrorism," says Mohamed ElBaradei, IAEA Director General, whose Agency sets world standards for nuclear safety and security. "We are not just dealing with the possibility of governments diverting nuclear materials into clandestine weapons programs. Now we have been alerted to the potential of terrorists targeting nuclear facilities or using radioactive sources to incite panic, contaminate property, and even cause injury or death among civilian populations."

"An unconventional threat requires an unconventional response, and the whole world needs to join together and take responsibility for the security of nuclear material," says Mr. ElBaradei. "Because radiation knows no frontiers, States need to recognise that safety and security of nuclear material is a legitimate concern of all States. Countries must demonstrate, not only to their own populations, but to their neighbours and the world that strong security systems are in place. The willingness of terrorists to commit suicide to achieve their evil aims makes the nuclear terrorism threat far more likely than it was before September 11."

The IAEA, the UN nuclear watchdog agency based in Vienna, helps countries around the world to prevent, intercept and respond to terrorist acts and other nuclear safety and security incidents. It has the only international response system in place that would be in a position to immediately react to assist countries in case of a radiological emergency caused by a nuclear terrorist attack.

Although terrorists have never used a nuclear weapon, reports that some terrorist groups, particularly al-Qaeda, have attempted to acquire nuclear material is a cause of great concern.

According to the IAEA, since 1993, there have been 175 cases of _____ and 201 cases of trafficking in other radioactive sources (medical, industrial). However, only 18 of these cases have actually involved small amounts of highly enriched uranium or plutonium, the material needed to produce a nuclear bomb. IAEA experts judge the quantities involved to be insufficient to construct a nuclear explosive device. "However, any such materials being in illicit commerce

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and conceivably accessible to terrorist groups is deeply troubling," says Mr. ElBaradei.

There has been a six-fold increase in nuclear material in peaceful programmes worldwide since 1970. According to IAEA figures, there are: 438 nuclear power reactors; 651 research reactors (of these 284 are in operation) and 250 fuel cycle plants around the world, including uranium mills and plants that convert, enrich store and reprocess nuclear material. Additionally, tens of thousands of radiation sources are used in medicine, industry, agriculture and research.

While the level of security at nuclear facilities is generally considered to be very high, security of medical and industrial radiation sources is disturbingly weak in some countries. "The controls on nuclear material and radioactive sources are uneven," says Mr. ElBaradei, "Security is as good as its weakest link and loose nuclear material in any country is a potential threat to the entire world."

The Risks Involved:

IAEA experts have evaluated the risks for nuclear terrorism in these three categories:

Nuclear facilities: IAEA experts believe the primary risks associated with nuclear facilities would involve the theft or diversion of nuclear material from the facility, or a physical attack or act of sabotage designed to cause an uncontrolled release of radioactivity to the surrounding environment.

From its inception, the nuclear industry has been keenly aware of the dangers of nuclear material falling into terrorist's hands. At all levels - operator, State and international - there is a complex infrastructure at work to ensure nuclear material is accounted for; safeguarded from diversion; and protected from theft and sabotage.

Billions of dollars per year are already being spent to protect and defend nuclear facilities. Indeed, no other industry in the world has such a sophisticated level of security. Nuclear facilities are protected by well-trained security forces and are extremely robust, designed to withstand, for example, earthquakes, tornado-force winds and accidental crashes of small aircraft. Although it is not automatic that any attack would result in a release of radioactivity, they are however industrial facilities and as such are not hardened to withstand acts of war.

The extent of damage that could be caused by the intentional crash of a large, fully fuelled jetliner into a nuclear reactor containment or other nuclear facilities is still a matter for analysis. Nuclear facility designs vary from country to country, so studies will have to take specific plant designs into account. "After September 11, we realized that nuclear facilities - like dams, refineries, chemical production facilities or skyscrapers - have their vulnerabilities," Mr. ElBaradei says. "There is no sanctuary anymore, no safety zone."

Countries around the world with nuclear facilities have heightened security since the 11 September attacks, and are conducting urgent analyses of their safety and security systems. The IAEA plans to strengthen and tailor its existing safety and security services to address the terrorism threat, by assisting countries in upgrading the security and safety of their nuclear facilities.

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Nuclear Material: According to IAEA experts, terrorists obtaining nuclear weapons would be the most devastating scenario. "While we cannot exclude the possibility that terrorists could get hold of some nuclear material," says Mr. ElBaradei, "it is highly unlikely they could use it to manufacture and successfully detonate a nuclear bomb. Still, no scenario is impossible."

Beyond the difficulty for terrorists to obtain weapon usable material - scientists estimate that 25 kg of highly enriched uranium or 8 kg of plutonium wound be needed make a bomb - actually producing a nuclear weapon is far from a trivial exercise. Scientific expertise and access to sophisticated equipment would be required. However, when the Cold War ended, thousands of highly knowledgeable scientists and engineers previously involved in the Soviet Union's weapons programme were laid off or found their incomes drastically reduced. Another legacy of the Cold War are the disturbing reports, albeit unsubstantiated, of missing nuclear weapons.

Nuclear material has traditionally been subjected to extensive national protection measures. To prevent theft of nuclear material, nuclear facilities employ a range of protection measures, including site security forces, site access control, employee screening and co-ordination with local and national security authorities. In some States, national security forces provide back-up to facility security. The IAEA offers countries around the world assessments and advice on physical security. It also maintains a database on incidents of , although the IAEA considers the information States provide on incidents and on follow-up to be inadequate.

In non-nuclear weapon States, the IAEA _____ to verify that nuclear material has not been diverted to non-peaceful uses. These safeguards, the verification tool entrusted to the IAEA in the 1970 Treaty on the Non-Proliferation of Nuclear Weapons (NPT), also play an important role in reducing the risk that terrorists could acquire nuclear material without detection. But when the NPT was drafted, nuclear terrorism was not perceived as a significant threat.

However, safeguards require that a state account for all its nuclear material material and serve as a "burglar alarm" against a terrorist. A well-designed system will also help to pinpoint the origin of missing material, identify individuals who had access to it, and facilitate recovery of the material.

The nuclear weapon programmes in the five Nuclear Weapon States - China, France, the Russian Federation, the United Kingdom and the United States, as well any that may exist in India, Pakistan and Israel, the three non-NPT countries known to have nuclear programmes - are not under the purview of IAEA safeguards. "Although I understand there is a high level of security for nuclear weapons," says Mr. ElBaradei, "I hope that all of these countries are urgently reviewing the safety and security of their nuclear weapons."

"There have been two nuclear shocks to the world already - the Chernobyl accident and the IAEA's discovery of Iraq's clandestine nuclear weapons programme," says Mr. ElBaradei. "It will be vital we do all in our power to prevent a third."

The IAEA plans to significantly expand its advisory services and help States upgrade protection of their nuclear materials.

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Radioactive Sources: IAEA experts are concerned that terrorists could develop a crude radiological dispersal device using radioactive sources commonly used in every day life. The number of radioactive sources around the world is vast: those used in radiotherapy alone are in the order of ten thousand. Many more are used in industry; for example, to check for welding errors or cracks in buildings, pipelines and structures. They are also used for the preservation of food. There is a large number of unwanted radioactive sources, many of them abandoned, others being simply "orphaned" of any regulatory control.

Such a weapon, sometimes referred to as a "dirty bomb", could be made by shrouding conventional explosives around a source containing radioactive material, although handling the nuclear material could well be deadly.

"Security of radioactive materials has traditionally been relatively light," says Abel Gonzalez, the IAEA's Director of Radiation and Waste Safety. "There are few security precautions on radiotherapy equipment and a large source could be removed quite easily, especially if those involved have no regard for their own health. Moreover, in many countries, the regulatory oversight of radiation sources is weak. As a result, an undetermined number of radioactive sources has become orphaned of regulatory control and their location is unknown."

"Certainly, the effects of a dirty bomb would not be devastating in terms of human life," says Mr. Gonzalez. "But contamination in even small quantities could have major psychological and economic effects."

The accidental contamination of Goiânia, a major city in Brazil, with a medical radiation source exemplifies the potential for a terrorist group to wreak havoc on an urban centre. In September 1987, scrap scavengers broke into an abandoned radiological clinic and stole a highly radioactive caesium 137 source and moved it to a junkyard for sale as scrap. Workers broke open the encasement and cut up the 20-gram capsule of caesium 137 into pieces. The valuable-looking scrap was then distributed to friends and family of workers around the city. Fourteen people were overexposed, and 249 contaminated. Four subsequently died. More than 110,000 people had to be continuously monitored. To decontaminate the area, 125,000 drums and 1470 boxes were filled with contaminated clothing, furniture, dirt and other materials; 85 houses had to be destroyed.

"We are dealing with a totally new equation since September 11," Mr. Gonzalez said. "These terrorists demonstrated before our eyes their willingness to give up their lives. The deadliness of handling intensely radioactive material can no longer be seen as an effective deterrent."

* * *

The IAEA is proposing a number of new initiatives, including strengthening border monitoring, helping States search for and dispose of orphan sources and strengthening the capabilities of the IAEA Emergency Response Centre to react to radiological emergencies following a terrorist attack.

"September 11 presented us with a clear and present danger and a global threat that requires global action," says Mohamed ElBaradei. "Many of our programs go to the heart of combating nuclear terrorism, but we now have to actively reinforce safeguards, expand our systems for combating smuggling in nuclear material and upgrade our safety and security

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services."

"At a minimum," Mr. ElBaradei says, "national assessments of security infrastructure for all types of nuclear and radioactive material should be required. Countries will have something to gain from allowing international assessments to demonstrate to the world that they are keeping their nuclear material secure."

In the short term, the IAEA estimates that at least $30-$50 million annually will be needed to strengthen and expand its programs to meet this terrorist threat.

* * *

The International Atomic Energy Agency (IAEA), based in Vienna, has 132 Member States. It has 2200 employees and an annual budget of about $330 Million. The IAEA, a UN agency, serves as the world's intergovernmental forum for scientific and technical co-operation in peaceful uses of nuclear energy. It is also the international inspectorate for the application of _____ measures to ensure that nuclear programmes are peaceful.

For further information, please contact Melissa Fleming, at (+43-1) 2600-21275; Mobile: (+43) 664-325 7376, E-mail: _____.

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News

Preparations Stepped Up For Possible New Attacks

Concern Focuses on Power Plants, Trucks, Ships, Bridges

By Eric Pianin and Dan Eggen
Washington Post Staff Writers
Thursday, November 1, 2001; Page A02

While authorities try to cope with the anthrax outbreak, federal and state officials are taking steps to prepare for a possible escalation of terrorism that experts say could include truck bombings and attacks on nuclear power plants as well as more hijackings.

Since the FBI issued its second national terrorism alert Monday, administration officials and congressional intelligence experts have studied myriad terrorist threats, including the outside possibility of the use of portable nuclear weapons. Steps taken by state and federal officials point, in particular, to concern about assaults on power plants and utilities, truck explosions in tunnels and on bridges, and attacks on ships carrying hazardous materials.

"If you're asking for a scenario of things that could go wrong, it's a mighty long list," said Rep. Porter J. Goss (R-Fla.), chairman of the House intelligence committee and a former CIA officer.

Yesterday, the governors of Arkansas, Louisiana and Mississippi ordered National Guard troops to strengthen

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(The Washington Post, Nov 1, 2001)
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security at nuclear facilities in their states, following a recommendation from Tom Ridge, the homeland security director, according to a spokesman for Entergy Corp. in Arkansas.

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A multimedia-based, geographic guide to the war on terrorism. *(Flash 5 required)*

In a conference call Tuesday, Ridge advised governors throughout the country to take such precautions if they had not already done so, according to Phil Fisher, the Entergy spokesman. The Federal Aviation Administration this week temporarily barred private aircraft from approaching 86 sensitive nuclear sites, including power plants and waste storage facilities.

The Treasury Department's Bureau of Alcohol, Tobacco and Firearms, meanwhile, has begun intensive inspections of all 9,500 mining and construction companies and others licensed to use explosives across the country. New York Gov. George E. Pataki (R) said that more than 1,500 National Guard troops patrolling in and around New York City will be armed for the first time by week's end.

Federal and local officials also remain concerned about the possibility that terrorists would attack ships carrying propane and other fuels. The city of Boston went to court in an attempt to keep liquefied natural gas tankers out of Boston harbor, but a judge ruled against the city on Monday -- just hours before the FBI issued its alert -- saying officials had failed to demonstrate a sufficient threat.

President Bush yesterday defended his decision to put the country on national security alert, telling business leaders that the country was still under attack.

"I wanted our law enforcement officials to know we had some information that made it necessary for us to protect United States assets, to protect those areas that might be vulnerable. And that's exactly what's taking place today," Bush said.

"This is a very unusual period in American history, obviously. We've never been attacked like this before. We're still being attacked," he said.

The nation has been awash in special warnings and alerts since Sept. 11.

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many focused on the types of potential terrorist targets that have been used in previous attacks or identified as possibilities by intelligence officials.

One example is commercial trucks, which have been used by terrorists around the world as delivery vehicles for makeshift but effective bombs.

Osama bin Laden's al Qaeda terrorist network has been particularly fond of explosives packed into trucks or cars, using the method in the first attack on the World Trade Center in 1993 and on the coordinated 1998 assaults on U.S. embassies in Kenya and Tanzania. A homegrown U.S. terrorist, Timothy J. McVeigh, used a rental truck to deliver the bomb that destroyed the Alfred P. Murrah Federal Building in Oklahoma City in 1995.

Since the Sept. 11 terrorist attacks on New York and Washington, the FBI and the Department of Transportation have warned the trucking industry to watch for suspicious activity in connection with hazardous chemicals, including radioactive waste and other substances that can be used to create weapons of mass destruction.

State and federal authorities in the United States have dramatically stepped up roadside inspections of tractor-trailers, especially those carrying hazardous materials, and Canadian officials are now asking for two forms of identification from truckers crossing the border, according to the American Trucking Associations industry group.

"We've been on high alert since September 11, and there is more focus on overall security in the industry," said Mike Russell, a spokesman for the trucking group. "We're transitioning to focus as much on security as on highway safety."

The ATF has temporarily halted its other regular inspections to focus on 9,500 mining and construction firms, fireworks factories and other companies that hold federal explosives licenses. The ATF is particularly interested in identifying any missing stocks, and has devoted a quarter of its agents to the task, an ATF official said.

The ATF and the FBI are still investigating the discovery of C-4 plastic explosive, along with a highly explosive, 1,000-foot strand of detonator cord, in a Greyhound bus locker in Philadelphia earlier last month. Authorities have determined that the cord was manufactured for military use, and have found no connection so far to the terrorist network blamed for the Sept. 11 attacks.

Nancy Savage, an FBI agent in Eugene, Ore., who is president of the FBI Agents Association, said the biggest concerns for investigators include airports, power plants and other key infrastructure points.

"Everyone expects additional attacks," Savage said. "We don't think

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they're going to give up now. That's why we're at war: We don't think they plan to give up anytime soon."

The FBI was particularly concerned in the weeks after Sept. 11 about crop-duster airplanes, which are fixtures in rural areas but which also could be used as part of a chemical or biological attack.

The presumed ringleader of the Sept. 11 hijackers, Mohamed Atta, showed interest in crop-dusters and how much poison they could carry, and even tried unsuccessfully this year to secure a U.S. government loan to purchase one. In addition, one of the key suspects now in U.S. custody, Zacarias Moussaoui, had information about crop-dusters on a computer.

The discoveries prompted the FAA to twice ground crop-dusters, and agricultural spraying companies have been asked to lock their planes and take other precautions since resuming flights.

Attorney General John D. Ashcroft said yesterday that the threat level announced Monday has not abated.

"I wish I could turn the clock back to before September the 11th," Ashcroft said. "I wish that we didn't have to talk about threats, I wish we didn't have to make announcements about threats. But the facts are different."

Staff writer Peter Behr contributed to this report.

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July 4, 2001

Terrorist Details His Training in Afghanistan

By LAURA MANSNERUS and JUDITH MILLER

An Algerian convicted of trying to carry out a terrorist attack in Los Angeles on the eve of the millennium celebration testified yesterday that he had received money and training at camps in Afghanistan that American officials say were run by Osama bin Laden.

The Associated Press

Ahmed Ressam testified on Tuesday against an Algerian former friend.

Ahmed Ressam, the Algerian, described in detail his training in light arms, explosives, assassinations and techniques for blowing up "the infrastructure of a country." After more than six months of training in Afghanistan in 1998, Mr. Ressam testified, he returned to Canada with $12,000 in seed money to plot terrorist attacks against the United States, Islam's "biggest enemy."

In his testimony yesterday in Federal District Court in Manhattan, Mr. Ressam did not mention Mr. bin Laden, the Saudi exile charged with conducting a jihad, or holy war, against the United States and its allies.

But in describing the origins of his plan to set off a bomb at Los Angeles International Airport, he nonetheless confirmed the key outlines of the picture drawn by American intelligence of Mr. bin Laden's operations. He described a network of camps in which Algerians, Jordanians, Germans and others were trained and indoctrinated for terrorist missions around the world.

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Mr. Ressam testified at the trial of Mokhtar Haouari, an Algerian accused of providing money and support for the plot to blow up the airport. In Los Angeles in April, Mr. Ressam was convicted of trying to bring explosives into the United States. He has since agreed to cooperate with prosecutors; his sentencing has been postponed to July 25.

Mr. Ressam's testimony, translated from Arabic by an interpreter, offered a rare insider's look at the design and attempted execution of a terrorist plot. His account depicted a decentralized structure in which militants were trained and given considerable latitude in selecting targets and missions.

In his testimony, Mr. Ressam said the camps were run by Abu Zubaida, the nom de guerre of a Palestinian whom American officials have identified as an important lieutenant to Mr. bin Laden.

American officials say Abu Zubaida reports directly to Mr. bin Laden and is in charge of recruiting for the camps. Mr. Ressam said Abu Zubaida arranged for his trip from Montreal to Afghanistan, providing him with Afghan clothes and an Afghan guide to take him from Pakistan to a camp called Khalden.

Mr. Ressam also described how, at the camps, he and others were made aware of orders to kill Americans that had been issued by Sheik Omar Abdel Rahman, the blind Egyptian cleric who was convicted in 1995 of conspiring to blow up the United Nations and other landmarks in the United States. He is now serving a life sentence in federal prison.

Mr. Ressam recounted how and why he selected the Los Angeles airport as a target and how he planned to rehearse and carry out the bombing. The plot went awry on Dec. 14, 1999, when a border guard in Port Angeles, Wash., questioned him in a routine check. Mr. Ressam, who does not speak English well, panicked and tried to flee. He was arrested and the authorities found more than 100 pounds of explosives in his car.

Mr. Ressam said he had planned the operation for more than a year but was forced to improvise when two other Algerians in his terrorist cell were detained in Britain and others then backed out. He said he selected the Los Angeles airport because he had passed through there on a flight from Pakistan.

Mr. Ressam testified that he wanted to test security at the airport by leaving a luggage cart with a bag unattended.

Mr. Ressam said he was trained at two camps in Afghanistan, Khalden and Darunta. Both have been identified by American officials as integral parts of al-Qaeda, a terrorist group founded by Mr. bin Laden that is an umbrella organization for anti-American militants around the world. There was no mention of al- Qaeda in the testimony yesterday, but Mr. Ressam was asked whether Abu Zubaida belonged to a "terrorist

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organization."

"Yes," he replied.

The United States has been pressing Afghanistan, most of which is ruled by the Taliban, to close down the camps and evict Mr. bin Laden. The Afghans have refused and American officials recently warned the Taliban that they would be held responsible for any attacks against the United States organized from their country.

A senior Bush administration official said Mr. Ressam's account "demonstrates that Afghanistan, in fact, has turned into the most threatening terrorist sanctuary in the world today."

Mr. Ressam said he was among 50 to 100 men at the camp in Afghanistan. He described his training in light weapons and explosives and instruction in "urban warfare." Among the possible targets among "enemies' installations," he said, were power plants, airports, railroads and large corporations.

Later, he said, he went to another camp for training in explosives, and returned to Canada with ingredients including hexamine, a booster used in bombs, and glycol. He said he bought other components in Vancouver and made his own timing devices.

When asked why he chose an airport as a target, he said, "An airport is sensitive politically and economically."

After Mr. Ressam had outlined his plan, he was asked if he realized that many civilians would die. "Yes, I would try to avoid that as much as possible," Mr. Ressam replied.

"But no matter how you did that, many would die," said Joseph F. Bianco, an assistant United States attorney.

"Yes," Mr. Ressam said.

Mr. Ressam is the star witness against Mr. Haouari, whom he met in Montreal through friends in a circle of Algerian émigrés. He agreed just a few weeks ago, as his sentencing date approached, to cooperate with the government.

Mr. Haouari is charged with providing money and support to Mr. Ressam, as well as bank fraud.

In testimony yesterday, Mr. Ressam, 34, began the story of a career that took him from a job in his father's coffee shop in Algeria to his arrest in 1999 with a cache of explosives in his rental car.

He described a life of petty crime in Montreal, where he arrived as an illegal immigrant in 1994 "to improve my life situation."

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"I lived on welfare and theft," he said. He said Mr. Haouari was dealing in stolen checks and passports and sometimes worked with him.

Mr. Ressam said that when he returned to Montreal from Afghanistan, he had been assigned to work with several other Algerians from the camps on general instructions to meet in Canada, rob banks and use the money to finance "an operation in America."

When his comrades failed to arrive in Canada, he testified, he worked mostly on his own.

He said that at the time, he and Mr. Haouari were working on a plan for Mr. Ressam to open a shop as a way to get information for counterfeit credit cards. He had told Mr. Haouari about the terrorist training camp, he said, and Mr. Haouari expressed interest in going, too.

He testified however, that he did not give Mr. Haouari details of the plan or identify the target.

"No, no, for security reasons I didn't want to tell him," Mr. Ressam said.

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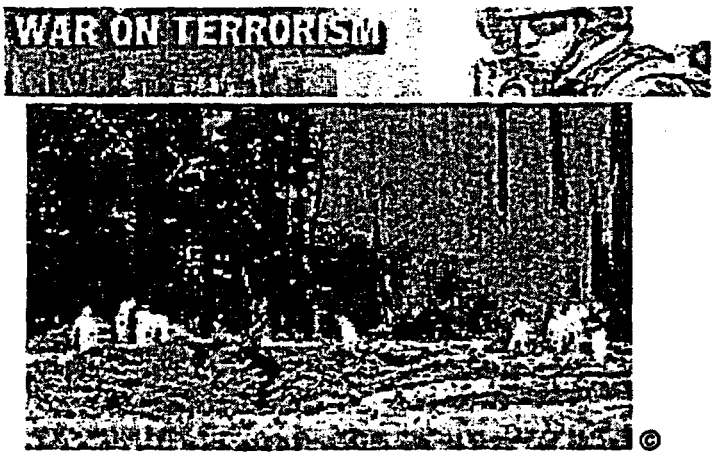

Mission aborted: the downed plane's flight path took it close to five nuclear plants
Photograph: Gary Tramontina

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WAR ON TERRORISM

Next: West Bank killings set stage for showdown

NUCLEAR MYSTERY: Crashed plane's target may have been reactor

Nicholas Rufford, David Leppard and Paul Eddy

THE hijackers who forced a fourth passenger jet to crash during the September 11 attacks in America may have been intending to use it to bomb a nuclear power station to cause a Chernobyl-type disaster.

The FBI is studying a report that the four terrorists who seized the plane may have been attempting to steer it towards a cluster of nuclear power stations on the east coast of America. The most likely target was Three Mile Island, site of America's most serious nuclear accident in 1979.

United Airlines flight 93 crashed into a field near the tiny town of Shanksville, in Pennsylvania, 90 minutes after taking off from Newark, New Jersey. All 44 passengers and crew on board died.

Until this weekend it had been assumed that the hijackers of the plane, a Boeing 757, were planning to fly it either to the presidential retreat at Camp David, or to Washington and crash it into the White House or the Congress and Senate buildings on Capitol Hill. But security officials have now revealed that within a week of the attacks, the FBI sent a report to MI5 saying that a "credible source" had said that the terrorists might have been planning to hit a nuclear plant.

Had it breached the plant's reactor vessel, such a strike could have caused an incident on the scale of the Chernobyl nuclear plant in Ukraine, which spread radioactive material over thousands of square miles in

1986.

US security sources say that Three Mile Island, which is part-owned by British Energy, was the subject of surveillance by some of the hijackers and their associates in the months before the terrorist attacks. One security official said: "Early on in the investigation we did receive a report from the FBI that the plane may have been heading for a nuclear power station. This was based on their analysis that Pittsburgh is near several power stations.

"There is some plausibility to this and we're not trying to dismiss it. But it may well be that nobody will ever know where the plane was going."

The "nuclear meltdown" assessment has not been independently confirmed but was taken seriously enough by the FBI to pass to European governments, including Britain and France.

The analysis is based on a study of flight 93's flight path and the fact that there are five nuclear power stations in the area. Experts say that the plane does not appear to have been hijacked until it was passing over West Virginia, some 200 miles beyond Washington. It then made a series of sharp turns before going into a steep descent. Aviation experts say that at this point there were three nuclear power stations between the plane and Washington and directly in its line of flight: Three Mile Island, Peach Bottom and Hope Creek.

Investigators cannot understand why the plane would have descended so early, unless its intended target was much nearer than Washington. The descent could have been an error by one of the hijackers, but if so, they cannot understand why the plane did not then climb again once control was regained.

America has since tightened security around nuclear stations and has taken steps to withdraw maps on the internet showing the location of nuclear plants. A French government minister said last week that fighters would shoot down aircraft heading for its nuclear plants. A missile defence system had been positioned at the Le Havre nuclear reprocessing plant.

In Britain, security around all nuclear sites has also been increased. David Blunkett, the home secretary, has given new powers to the 500-strong police force that guards the sites. Atomic Energy Authority police will be able to patrol an extra 13 civil nuclear sites, including Sizewell, Hinkley Point and Dungeness.

Engineering experts are divided over whether concrete containment shields around nuclear power stations could withstand a direct hit from a large passenger aircraft, especially one carrying 200,000lb of fuel, as was flight 93, enough to reach its destination of San Francisco.

The containment buildings generally have an outer structure, which for much of the dome is 3ft-thick concrete containing large amounts of reinforcing steel. Inside is a steel "lining" 1in-4in thick.

There are usually two more concrete walls close to the reactor, each 1ft thick and with reinforced steel bars. But these walls do not enclose the top of the reactor completely. The reactor vessel itself is about 4in-6in thick and made of high-carbon steel.

All reactors are designed to withstand impact by a light plane. Experts say it is unclear whether a larger modern jet loaded with fuel, deliberately flown at high speed, could break open the reactor vessel. The resultant fire could, however, cause enough damage to allow radioactive material into the air.

The drama aboard flight 93 as a small group of passengers tried to seize control of the plane from the hijackers during its final few minutes has become an emblem of American heroism during the events of September 11.

Delayed 40 minutes in taking off from Newark's congested airport, the plane was in the early stages of its journey when its passengers started hearing that other aircraft had been hijacked and at least one had flown into the twin towers of the World Trade Center.

Todd Beamer, one of the passengers, called an emergency operator on an onboard telephone after he and fellow passengers learnt of the first attack. He explained that flight 93 had also been hijacked. He said there were three hijackers - two with knives and one with what he thought was a bomb strapped to his waist. In fact, there were four, and by this time the fourth was almost certainly flying the plane.

Beamer, who was married with two young sons, told the operator: "We're going to do something. I know I'm not going to get out of this." He explained that some of passengers had decided to jump on the terrorist thought to have the bomb.

With the telephone left on, he could be heard saying: "Are you guys ready? Let's roll." The operator heard screams and a few minutes later the line went dead.

- The FBI is looking into whether another United Airlines flight, scheduled to leave Kennedy International Airport for San Francisco, was a target of hijackers on September 11. When the plane was grounded because of the attacks, four Middle Eastern-looking men refused to return to their seats and hurriedly left as soon as its doors opened.

Next page: West Bank killings set stage for showdown

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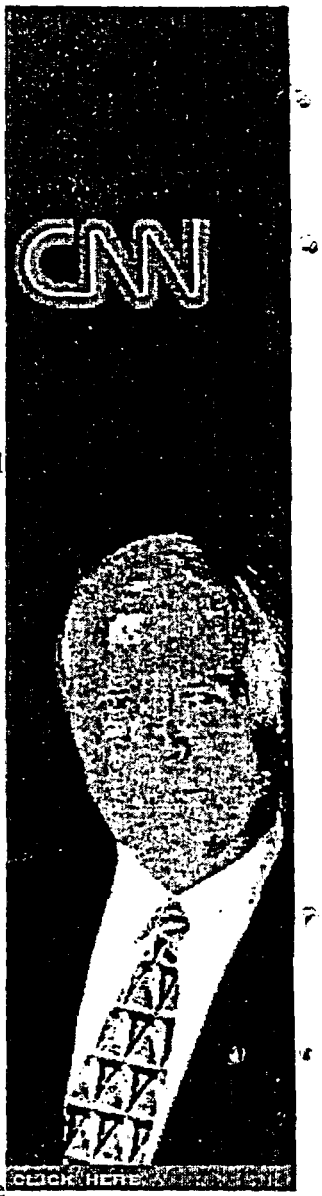

FAA restricts flights near World Series, nuclear plants

October 30, 2001 Posted: 10:41 p.m. EDT (0241 GMT)

(AP PHOTO)

President Bush throws out the ceremonial first pitch before Game 3 of the World Series at New York's Yankee Stadium.

WASHINGTON (CNN) -- Heeding the latest terrorist warning, the Federal Aviation Administration is temporarily restricting flights near the World Series games in New York and around nuclear sites.

The restrictions over New York City prohibit any aircraft operating under visual flight rules from flying within 30 nautical miles of John F. Kennedy International Airport during World Series games. The ban is in effect from 6:45 p.m. until 2 a.m. ET and lasts until midnight, November 6.

Restrictions were even tighter during President Bush's appearance at Tuesday's game.

All aircraft flying below 3,000 feet were prohibited from approaching within three nautical miles of JFK airport from 7:05 p.m. until 7:15 p.m. ET, and from 10:30 p.m. until 10:45 p.m. ET.

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The nuclear sites ban, which affects 80 facilities such as power plants and Energy Department areas, restricts aircraft flying below 18,000 feet from coming within a radius of 10 nautical miles of each facility.

The restrictions underscore Attorney General John Ashcroft's announcement Monday of "credible reports" that another major terrorist attack may be possible within the week.

"The FAA realizes these restrictions inconvenience general aviation pilots and airports," said FAA Administrator Jane Garvey in a press release. "As the FAA and other federal agencies continuously review measures to ensure national security, we look for the understanding and cooperation of the general aviation community."

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"Pilots must make every effort ... to avoid these sensitive areas," said Phil Boyer, president of the Aircraft Owners and Pilots Association.

Elsewhere in Washington, officials said Tuesday that Bush's national security team wants the war in Afghanistan to continue during Ramadan, the Muslim holy period. Letting up on the assaults, said one official, would be interpreted "as a sign of flexibility in what the president himself has called a doctrine."

The president, officials said, would decide whether to press on during Ramadan, which begins about November 17.

The sentiment to continue strikes during Ramadan underscored Army Gen. Tommy Franks' assertion Tuesday that the allied coalition is committed "for as long as it takes" to oust the ruling Taliban and Osama bin Laden's al Qaeda network.

After meeting with Uzbekistan's president, Franks, commander of the U.S. military campaign in Afghanistan, dismissed suggestions that the U.S-led military operation has bogged down in its fourth week. Full story)

U.S. airplanes launched another round of airstrikes early Tuesday on the Taliban stronghold of Kandahar, with low-flying jets pounding targets around the southern Afghan city and Taliban forces returning antiaircraft fire.

CNN's Kamal Hyder reported Kandahar was quiet Tuesday night, but electricity was out to most of the city and there was no running water. (full story)

Ashcroft warned Monday that there is a "credible" threat of new terrorist action in the United States in the coming days, and he told law enforcement agencies and the public to be on "highest alert."

The attorney general said intelligence sources had found nonspecific but "credible" information the nation could be the focus again for some sort of terrorist attack.

"We are dealing with an unknown; we are dealing without a lot of specific information," Homeland Security Director Tom Ridge told CNN on Tuesday. "But we also know since September 11 the environment is different, and America has to continue to be on guard."

Latest developments

• Homeland Security Director Tom Ridge said Tuesday that the security alert issued Monday by Attorney General John Ashcroft was a "reiteration" of the October 11 alarm sent to law enforcement agencies. Ashcroft warned that there is a "credible" threat of new terrorist action in the United States in the coming days, and he told law enforcement agencies and the public to be on "highest alert."

U.S. law enforcement is on high alert after officials warn of possible terrorist attacks. CNN's John King reports (October 30)

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• Despite the new threat of more terror attacks in

the United States, President Bush plans to be at Yankee Stadium in New York on Tuesday night to throw out the first pitch in the third game of the World Series. (Full story)

• Transportation Secretary Norman Mineta announced Tuesday a new crackdown on security at U.S. airports, including more stringent passenger and bag searches. Mineta said there have been deficiencies in security screening since September 11, and the aviation industry must show improvement "right away." (Full story)

• American Airlines, the world's largest airline, announced Tuesday that it has secured the cockpits of its entire fleet of jet aircraft. American said the new locking devices prevent intruders from accessing the cockpit by securing it from the inside.

• Amid growing criticism of the U.S.-led military campaign, British Prime Minister Tony Blair appealed to the public Tuesday to remember the images of September 11. "Thousands of people were killed in cold blood in the worst terrorist attacks the world has ever seen," Blair said in a keynote speech. (Full story)

• An Egyptian-born man was charged Tuesday with conspiring with others to murder Northern Alliance leader Gen. Ahmad Shah Masood. The opposition leader was killed September 9 by suicide bombers posing as journalists. Yasser Al Siri, who has lived in Britain for eight years, was charged under Britain's anti-terrorism act. (Full story)

• Iraqi President Saddam Hussein warns that the U.S.-led war in Afghanistan could spread. Calling it a spark that may set "the world on fire," he called Tuesday on other countries to help defeat the United States. (Full story)

• Vice President Dick Cheney on Tuesday was placed again at an undisclosed, secure location, administration officials told CNN. They said Cheney -- who made public appearances Monday in New York-- was returned to the higher level of security through at least next week in response to the latest warning that Ashcroft announced.

• The United Nations refugee agency said Tuesday that thousands of Afghans continue to cross into Pakistan through unofficial entry points. At a briefing in Geneva, Switzerland, a U.N. spokesman said that more than 5,000 people

The Pentagon says progress is being made in the military campaign in Afghanistan. CNN's Jamie McIntyre reports (October 30)

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Will Americans continue to take alerts seriously if attacks don't follow?

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crossed into Pakistan's Northwest Frontier province last weekend.

• The U.N. special envoy to Afghanistan, Lakhdar Brahimi, was meeting Tuesday with Pakistan's President Pervez Musharraf and met with other top Pakistani officials Monday as part of the U.N.'s role in shaping a postTaliban Afghanistan. Brahimi said the United Nations is considering a proposal by the Taliban ambassador in Pakistan to discuss the situation in Afghanistan.

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In Afghan Jail, a Terrorist Who Won't Surrender

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By William Branigin
Washington Post Staff Writer
Tuesday, October 30, 2001; Page A13

KHOJA MAKBUL, Afghanistan -- He sits cross-legged on a carpet, fixing his visitor intently with dark eyes behind thick, oversize glasses. With his flowing black beard, embroidered skullcap and beatific smile, the soft-spoken Pakistani still resembles the Islamic scholar he once was.

But, by his own account, Salahuddin Khaled is a dangerous man. The 27-year-old member of the hard-line Pakistani Muslim group Harkat ul-Mujaheddin is an ally of Afghanistan's ruling Taliban militia, a disciple of Osama bin Laden and a highly trained terrorist. After five years behind bars, he is the longest serving prisoner of war held by the opposition Northern Alliance. And there is a reason he is likely to continue to serve time in a single-story, mud-brick compound in the Panjshir Valley.

If released, he readily acknowledges, he would gladly carry out the kind of terrorist attacks that killed almost 5,000 people in the United States last month.

He talks of using atomic weapons against America, and wonders

AMERICA AT WAR
SPECIAL REPORT

(Reuters, Oct 30, 2001)

(The Washington Post, Oct 30, 2001)

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whether the Sept. 11 attackers would have better served their cause by flying one of the hijacked planes into a nuclear power plant. "I don't know who did that action," Khaled says in halting English. "If Muslim organizations did that action, I agree, because America is their enemy. . . . They have to hurt America in its military, economic and political centers to make America leave its plans against Islam."

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While it may never be possible to fully understand the passions and motives of someone such as Khaled, an interview with him at the Northern Alliance's Baharak Prison provides a glimpse into the mind of a terrorist. Never raising his voice, he explains in his rudimentary English and fluent Dari, the language of the Tajik ethnic group here, the reasons for his implacable hatred of the United States.

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He occasionally expresses regret for the deaths of civilians in the Sept. 11 attacks and other operations against the United States. But he makes it clear that these casualties are secondary to the goal of punishing America for a long list of sins, headed by U.S. support for Israel.

He also makes it clear that he sees the United States as the main obstacle to establishing Islamic law in Afghanistan, across Central Asia and elsewhere around the world.

Khaled is among 21 foreigners and 306 Afghan Taliban members held at the Baharak Prison, which opened eight months ago in a desolate spot a couple of miles north of the village of Baharak. It is reached by leaving a dirt road cut into a mountainside, fording a shallow part of the Panjshir River and walking across a narrow metal bridge. The prison sits on a rocky spit of land that juts into a bend in the river and is bordered by barren, forbidding mountains that rise almost vertically from the riverbed.

The fortress-like prison contains a row of dark cells facing a sunny interior courtyard. Each cell is about 12 feet wide by 28 feet long and crammed with more than 30 inmates. Shoes, bags of clothes and other belongings hang from the walls and from log beams in the ceiling.

The other foreign prisoners are from Burma, China, Yemen and Iraq, as well as Pakistan. Like Khaled, they joined extremist Muslim organizations that urged them to fight for the Taliban in its "holy war" in Afghanistan.

Abdul Jabar, 22, a Pakistani with a slight build and a sparse beard, studied at a veterinary college for two years before coming to Afghanistan in 1999. After three weeks of training, he was sent to the front to fight the Northern Alliance and was promptly captured.

Interviewed in a prison sitting room, he said he still believes in the Taliban cause, admires bin Laden and hates the U.S. government "because it helps Israel." He said he came to Afghanistan believing that he was going to be fighting Russians, but found himself facing only Muslim Afghans. (The Soviet Union withdrew its occupation forces from Afghanistan in 1989.) If he had known that, he said, "I would not fight. I would fight against Israel or in Kashmir or against America."

Would he carry out a suicide bombing? "If I know that [the target] is not Muslim but is Jewish or [of] another religion, I would immediately carry out this action," he said. But if he knew the target were Muslim, he would question the order.

Noor Mohammed Abdullah, 29, a Muslim from China, was also captured two years ago after barely a month in Afghanistan. He came here from a Pakistani Islamic school whose principal has close ties to the Taliban and told students they had an obligation to fight in Afghanistan, he said. He said he was told he would be fighting Russians and Americans.

Abdullah said he now realizes he made a mistake. If released, he would continue his religious studies and no longer fight the Northern Alliance, "because they are Muslim," he said.

Khaled has no such qualms.

"He will not change his ideas," said Abdul Qayyum, an Afghan with gray hair and pale blue eyes who is a deputy warden of the prison.

Another deputy warden, Farouk, said Khaled is the hardest of the hard-liners at Baharak, and probably the most dangerous man in the prison. "Any time he is free, he will work with Osama bin Laden and the Taliban," Farouk said.

Khaled has consistently expressed his radical views to all who will

listen, seemingly unconcerned about the chilling effect his words have on any prospect of release or a prisoner exchange. It is as if softening his hard-line positions would be tantamount to renouncing his faith, Farouk said.

At one time, the Northern Alliance held 2,500 Taliban prisoners. Most have since been exchanged for alliance POWs.

In a study this year on foreign prisoners held by the Northern Alliance, Julie Sirrs, a former Afghan analyst for the Defense Intelligence Agency, reported that more non-Afghans are fighting in the country than ever before. She said the foreigners are more difficult to capture because they tend to be more motivated than Afghans on the Taliban side, "many of whom are conscripts."

Contrary to the popular perception, Sirrs said, only 43 percent of the 113 prisoners she interviewed identified themselves as *talibs*, or religious students. Most of the foreign fighters were recruited while working as shopkeepers, laborers, party activists and in other occupations. Only 30 percent of the Pakistanis identified themselves as Pashtuns, the ethnic group that dominates the Taliban.

Khaled said he was born in Baluchistan province, the son of an Islamic studies professor. He earned a degree in Islamic law from the Islamia University of Punjab, where friends recruited him into the fundamentalist Harkat ul-Mujaheddin, which has been implicated in the deaths of Americans and declared a terrorist organization by the U.S. government. Khaled arrived in Afghanistan in 1992 determined, he said, to "defend Muslims" at a time when Serbs were killing Muslims in Bosnia.

He said he underwent training for two years near Khost, south of the capital, Kabul, first in a Harkat camp. He later moved up to a camp for more advanced trainees that eventually was taken over by bin Laden, he said. There, 35 men received instruction in guerrilla tactics, bomb-making and "chemicals and poisons." He said the last instruction covered "poison gas and bombs," but he declined to go into details. He denied any knowledge of anthrax, the disease currently being spread in the United States by biological agents sent through the mail, and said its use was not taught while he was at the training center.

After bin Laden moved to Afghanistan in 1995, he occasionally visited the camp to give pep talks, said Khaled, who was serving as an instructor by then. Bin Laden would tell the trainees, "You should spill more sweat during training so you don't spill your blood during battle," Khaled recalled. He said bin Laden told the students they were "fighting against people who want to finish Islam" and that the U.S. government was an enemy of Muslims.

After his training, Khaled said he went to Kashmir to fight the Indian

government, returning a year later to Afghanistan, where he linked up with the Taliban.

He was in command of 30 Harkat fighters when his unit was cut off by Northern Alliance soldiers near Jabal Saraj in October 1996. While his men escaped to Kabul, he said he held off his attackers for five hours, using an AK-47 assault rifle and the sharpshooting skills he acquired during his training. He said he killed more than 10 alliance fighters before he ran out of ammunition and surrendered.

Today, Khaled is as unapologetic about his support for the Taliban as he is about his hatred for America.

"If America didn't work against us, we would never take action against their cities," he said. "Our enemies are America and Israel, but we cannot fight against them face to face. We have to fight against them secretly to make them leave their plans and stop working against us."

Besides support for Israel, including $3 billion a year in aid, Khaled's catalogue of complaints against the United States includes the activities of the CIA, the presence of U.S. troops on "holy land" in Saudi Arabia since the Persian Gulf War (a major grievance of bin Laden), the U.S. intervention in Somalia, the bombing of Iraq, an airstrike against Libya, the dropping of atomic bombs on Japan during World War II and even Washington's rejection of the Kyoto treaty on global warming.

At one point, he launched into a stream of revisionist history, portraying the United States as the aggressor against Japan in World War II and justifying the attack on Pearl Harbor. Hawaii, he asserted, rightfully belonged to Japan.

Of the Sept. 11 attacks, Khaled said. "the target was not to kill civilian people, but [to cause] important hurt to the American government." There was no other way to achieve the goal, he said. "They had to do that action."

Would he participate in such a mission? "If my commander led me to an action like this, I would do it. If Osama bin Laden told me to do it, I would do it." He added, "I'm from Harkat, but I think that Osama bin Laden is also my leader."

He has no reservations about sacrificing himself in a suicide attack, he said, "because our target is very important, more important than my life." But he said he did not necessarily agree with the specific targets last month.

"In America. there are more important places, like atomic plants and reactors [that] they could attack," he said. "Not only atomic plants, but the CIA center. arms factories and the White House."

"America tries to say to the world that Muslim fundamentalist organizations [are] terrorist," Khaled said. "But we think the American government is terrorist."

He rejected the notion that the terrorist attacks last month were counterproductive, since they did not cause the United States to reconsider its policies but united Americans in a desire for revenge and brought an intervention in Afghanistan aimed at eliminating bin Laden and the Taliban.

"You will see, America will not be successful in [its] goals," Khaled said. The use of U.S. troops will eventually be necessary, he said, and "American ground forces wouldn't be able to fight against us. We're fighting the Americans on three front lines: in Afghanistan, in Pakistan and in the United States."

Even if bin Laden is killed, Khaled said, another leader will quickly take his place. And if the Taliban eventually is defeated, "we will do secret activities, secret action."

Does he mean more terrorist attacks?

Khaled smiled broadly. "That's right."

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NRC NEWS

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No. 01-112 September 21, 2001

NRC REACTS TO TERRORIST ATTACKS

In light of the recent terrorist attacks, U.S. Nuclear Regulatory Commission officials and staff have been working around the clock to ensure adequate protection of nuclear power plants and nuclear fuel facilities. This has involved close coordination with the Federal Bureau of Investigation, other intelligence and law enforcement agencies, NRC licensees, and military, state and local authorities.

Immediately after the attacks, the NRC advised nuclear power plants to go to the highest level of security, which they promptly did. The NRC has advised its licensees to maintain heightened security. The agency continues to monitor the situation, and is prepared to make any adjustments to security measures as may be deemed appropriate.

In view of the recent unprecedented events, Chairman Richard A. Meserve, with the full support of the Commission, has directed the staff to review the NRC's security regulations and procedures.

A number of questions have come in from reporters and members of the public since the tragic events of September 11. The following questions and answers are offered in response:

Q: What would happen if a large commercial airliner was intentionally crashed into a nuclear power plant?

A:. Nuclear power plants have inherent capability to protect public health and safety through such features as robust containment buildings, redundant safety systems, and highly trained operators. They are among the most hardened structures in the country and are designed to withstand extreme events, such as hurricanes, tornadoes and earthquakes. In addition, all NRC licenses with significant radiological material have emergency response plans to enable the mitigation of impacts on the public in the event of a release. However, the NRC did not specifically contemplate attacks by aircraft such as Boeing 757s or 767s and nuclear power plants were not designed to withstand such crashes. Detailed engineering analyses of a large airliner crash have not yet been performed.

Q: What measures have the NRC and its power plant licensees taken in face of this potential threat?

A: Immediately after the attacks, the NRC advised licensees to go to the highest level of security, which all did promptly. The specific actions are understandably sensitive, but they generally included such things as increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with law enforcement and military authorities, and limited access of personnel and vehicles to the sites.

Q: What, precisely, did the NRC do in response to the attacks?

A: At 10 a.m. on September 11, the NRC activated its Emergency Operations Center in headquarters and assembled a team of top officials and specialists. The same was done in each of its four regional offices. In addition to communicating with its licensees about the need to go to the highest level of security, the NRC established communications with the FBI, the Department of Energy, and the Federal Emergency Management Agency, among others. NRC personnel were dispatched to the FBI's Strategic Information Operations Center. The NRC has also established close communications with nuclear regulators in Canada and Mexico.

Q:What would happen if a large aircraft should crash into a spent fuel dry storage cask?

A: The capacity of spent fuel dry storage casks to withstand a crash by a large commercial aircraft has not been analyzed. Nonetheless, storage casks are robust and must be capable of withstanding severe impacts, such as might occur during tornadoes, hurricanes or earthquakes. In the event that a cask were breached, any impacts would be localized. All spent fuel storage facilities have plans to respond to such an emergency, drawn up in consultation with local officials.

Q: What if a large aircraft crashed into a spent fuel transportation cask in a heavily populated area?

A: Again, the capacity of shipping casks to withstand such a crash has not been analyzed. However, they are designed to protect the public in severe transportation accidents. The cask must be able to withstand a 30-foot drop puncture test, exposure to a 30-minute fire at 1475 degrees Fahrenheit, and submersion under water for an extended period. Moreover, the location of loaded casks is not publicly disclosed and such a cask would present a small target to an aircraft .

If an airliner crashed into a cask, there could be some localized impacts. Regulations require special accident response training of those involved in shipping, as well as coordination with state, local and tribal emergency response personnel. In addition, redundant communications must be maintained during shipment with the transporter vehicle; this would facilitate emergency response, if necessary.

Q: Could such a crash into a nuclear power plant, or a storage or shipping cask trigger a nuclear explosion?

A: No.

Q: What are the consequences if an airliner crashed into a uranium fuel cycle facility?

A: Because of the nature of the material, there would likely be only minimal off-site radiological consequences. Some such facilities use chemicals similar to those found at many industrial facilities. In the event of a release, comprehensive emergency response procedures would be immediately implemented.

Q: Have nuclear power plants been subject to attack in the past?

A: There has never been an attack on a nuclear power plant. On very rare occasions there have been intrusions. For example, there was a 1993 car crash through the gates of Three Mile Island plant by an individual with a history of treatment for mental illness. Such intrusions have not resulted in harm to public health or safety.

Q: What are the normal security measures at commercial nuclear power plants.

A: Licensees are required to implement security programs that include well-armed civilian guard forces, physical barriers, detection systems, access controls, alarm stations, and detailed response strategies. NRC routinely inspects security measures as part of its normal reactor oversight process and periodically undertakes various exercises, including force-on-force exercises, so as to assure that any vulnerabilities are exposed and corrected .

Q: Is an attack using an airplane part of the NRC's design basis threat against which its licensees have to defend?

A: No. The NRC has been in close and continuing contact with law enforcement and the military regarding such a threat.

Q: What exactly is the so-called design basis threat?

A: The details of the design basis threat are classified, but it includes the characteristics of a possible sabotage attempt that NRC licensees are required to protect against. The agency continually assesses the adequacy of the design basis threat in consultation with local law enforcement and federal intelligence agencies.

Q:Is the NRC contemplating a modification of the design basis threat?

A: The agency will continue to coordinate with law enforcement and intelligence agencies to assess the implications of this new manifestation of terrorism. If the NRC determines that the design basis threat warrants revision, such changes would occur through a public rulemaking.

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Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants

October 2000

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Technical Study of Spent Fuel Pool Accidents at Decommissioning Plants

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EXECUTIVE SUMMARY

This report documents a study of spent fuel pool (SFP) accident risk at decommissioning nuclear power plants. The study was undertaken to support development of a risk-informed technical basis for reviewing exemption requests and a regulatory framework for integrated rulemaking.

The staff published a draft study in February 2000 for public comment and significant comments were received from the public and the Advisory Committee on Reactor Safeguards (ACRS). To address these comments the staff did further analyses and also added sensitivity studies on evacuation timing to assess the risk significance of relaxed offsite emergency preparedness requirements during decommissioning. The staff based its sensitivity assessment on the guidance in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis." The staff's analyses and conclusions apply to decommissioning facilities with SFPs that meet the design and operational characteristics assumed in the risk analysis. These characteristics are identified in the study as industry decommissioning commitments (IDCs) and staff decommissioning assumptions (SDAs). Provisions for confirmation of these characteristics would need to be an integral part of rulemaking.

The results of the study indicate that the risk at SFPs is low and well within the Commission's Quantitative Health Objectives (QHOs). The risk is low because of the very low likelihood of a zirconium fire even though the consequences from a zirconium fire could be serious. The results are shown in Figures ES-1 and ES-2. Because of the importance of seismic events in the analysis, and the considerable uncertainty in seismic hazard estimates, the results are presented for both the Lawrence Livermore National Laboratory (LLNL) and the Electric Power Research Institute (EPRI) seismic hazard estimates. In addition, to address a concern raised by the ACRS, the results also include a sensitivity to a large ruthenium and fuel fines release fraction. As illustrated in the figures, the risk is well below the QHOs for both the individual risk of early fatality and the individual risk of latent cancer fatality.

The study includes use of a pool performance guideline (PPG) as an indicator of low risk at decommissioning facilities. The recommended PPG value for events leading to uncovery of the spent fuel was based on similarities in the consequences from a SFP zirconium fire to the consequences from a large early release event at an operating reactor. A value equal to the large early release frequency (LERF) criterion (1×10^{-5} per year) was recommended for the PPG. By maintaining the frequency of events leading to uncovery of the spent fuel at decommissioning facilities below the PPG, the risk from zirconium fires will be low and consistent with the guidance in RG 1.174 for allowing changes to the plant licensing basis that slightly increase risk. With one exception (the H.B. Robinson site) all Central and Eastern sites which implement the IDCs and SDAs would be expected to meet the PPG regardless of whether LLNL or EPRI seismic hazard estimates are assumed. The Robinson site would satisfy the PPG if the EPRI hazard estimate is applied but not if the LLNL hazard is used. Therefore, Western sites and Robinson would need to be considered on a site-specific basis because of important differences in seismically induced failure potential of the SFPs.

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The appropriateness of the PPG was questioned by the ACRS in view of potential effects of the fission product ruthenium, the release of fuel fines, and the effects of revised plume parameters. The staff added sensitivity studies to its analyses to examine these issues. The consequences of a significant release of ruthenium and fuel fines were found to be notable, but not so important as to render inappropriate the staff's proposed PPG of $1x10^{-5}$ per year. The plume parameter sensitivities were found to be of lesser significance.

In its thermal-hydraulic analysis, documented in Appendix 1A, the staff concluded that it was not feasible, without numerous constraints, to establish a generic decay heat level (and therefore a decay time) beyond which a zirconium fire is physically impossible. Heat removal is very sensitive to these additional constraints, which involve factors such as fuel assembly geometry and SFP rack configuration. However, fuel assembly geometry and rack configuration are plant specific, and both are subject to unpredictable changes after an earthquake or cask drop that drains the pool. Therefore, since a non-negligible decay heat source lasts many years and since configurations ensuring sufficient air flow for cooling cannot be assured, the possibility of reaching the zirconium ignition temperature cannot be precluded on a generic basis.

The staff found that the event sequences important to risk at decommissioning plants are limited to large earthquakes and cask drop events. For emergency planning (EP) assessments this is an important difference relative to operating plants where typically a large number of different sequences make significant contributions to risk. Relaxation of offsite EP a few months after shutdown resulted in only a "small change" in risk, consistent with the guidance of RG 1.174. Figures ES-1 and ES-2 illustrate this finding. The change in risk due to relaxation of offsite EP is small because the overall risk is low, and because even under current EP requirements, EP was judged to have marginal impact on evacuation effectiveness in the severe earthquakes that dominate SFP risk. All other sequences including cask drops (for which emergency planning is expected to be more effective) are too low in likelihood to have a significant impact on risk. For comparison, at operating reactors additional risk-significant accidents for which EP is expected to provide dose savings are on the order of $1x10^{-5}$ per year, while for decommissioning facilities, the largest contributor for which EP would provide dose savings is about two orders of magnitude lower (cask drop sequence at $2x10^{-7}$ per year).[1] Other policy considerations beyond the scope of this technical study will need to be considered for EP requirement revisions and previous exemptions because a criteria of sufficient cooling to preclude a fire cannot be satisfied on a generic basis.

Insurance does not lend itself to a "small change in risk" analysis because insurance affects neither the probability nor the consequences of an event. As seen in figure ES-2, as long as a zirconium fire is possible, the long-term consequences of an SFP fire may be significant. These long-term consequences (and risk) decrease very slowly because cesium-137 has a half life of approximately 30 years. The thermal-hydraulic analysis indicates that when air flow has been restricted, such as might occur after a cask drop or major earthquake, the possibility of a fire lasts many years and a criterion of "sufficient cooling to preclude a fire" can not be defined on a

[1]Consistent with PRA limitations and practice, contributions to risk from safeguards events are not included in these frequency estimates. EP might also provide dose savings in such events.

generic basis. Other policy considerations beyond the scope of this technical study will therefore need to be considered for insurance requirements.

The study also discusses implications for security provisions at decommissioning plants. For security, risk insights can be used to determine what targets are important to protect against sabotage. However, any revisions in security provisions should be constrained by an effectiveness assessment of the safeguards provisions against a design-basis threat. Because the possibility of a zirconium fire leading to a large fission product release cannot be ruled out even many years after final shutdown, the safeguards provisions at decommissioning plants should undergo further review. The results of this study may have implications on previous exemptions at decommissioning sites, devitalization of spent fuel pools at operating reactors and related regulatory activities.

The staff's risk analyses were complicated by a lack of data on severe-earthquake return frequencies, source term generation in an air environment, and SFP design variability. Although the staff believes that decommissioning rulemaking can proceed on the basis of the current assessment, more research may be useful to reduce uncertainties and to provide insights on operating reactor safety. In particular, the staff believes that research may be useful on source term generation in air, which could also be important to the risk of accidents at operating reactors during shutdowns, when the reactor coolant system and the primary containment may both be open.

In summary, the study finds that:

1. The risk at decommissioning plants is low and well within the Commission's safety goals. The risk is low because of the very low likelihood of a zirconium fire even though the consequences from a zirconium fire could be serious.

2. The overall low risk in conjunction with important differences in dominant sequences relative to operating reactors, results in a small change in risk at decommissioning plants if offsite emergency planning is relaxed. The change is consistent with staff guidelines for small increases in risk.

3. Insurance, security, and emergency planning requirement revisions need to be considered in light of other policy considerations, because a criterion of "sufficient cooling to preclude a fire" cannot be satisfied on a generic basis.

4. Research on source term generation in an air environment would be useful for reducing uncertainties.

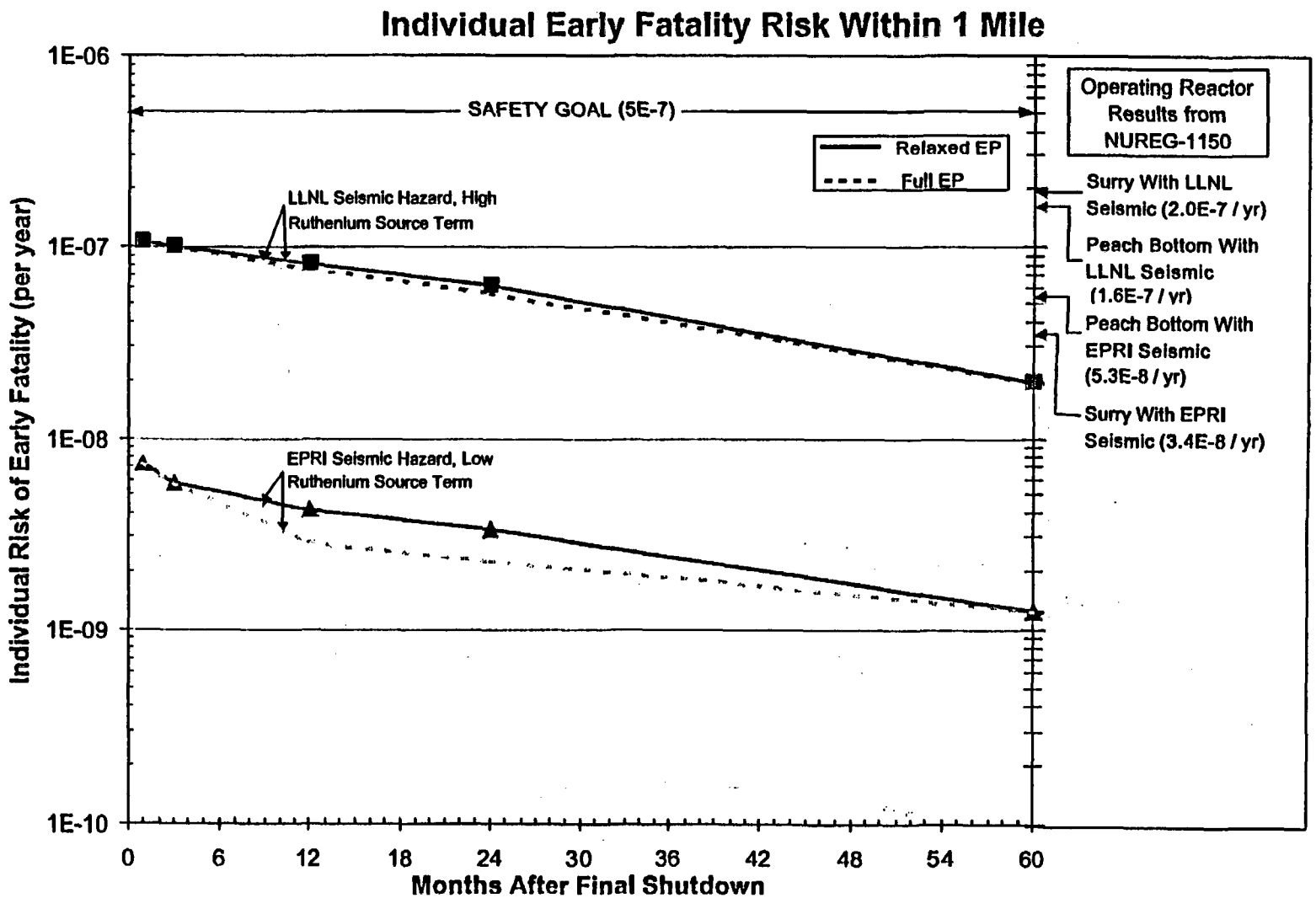

Figure ES-1

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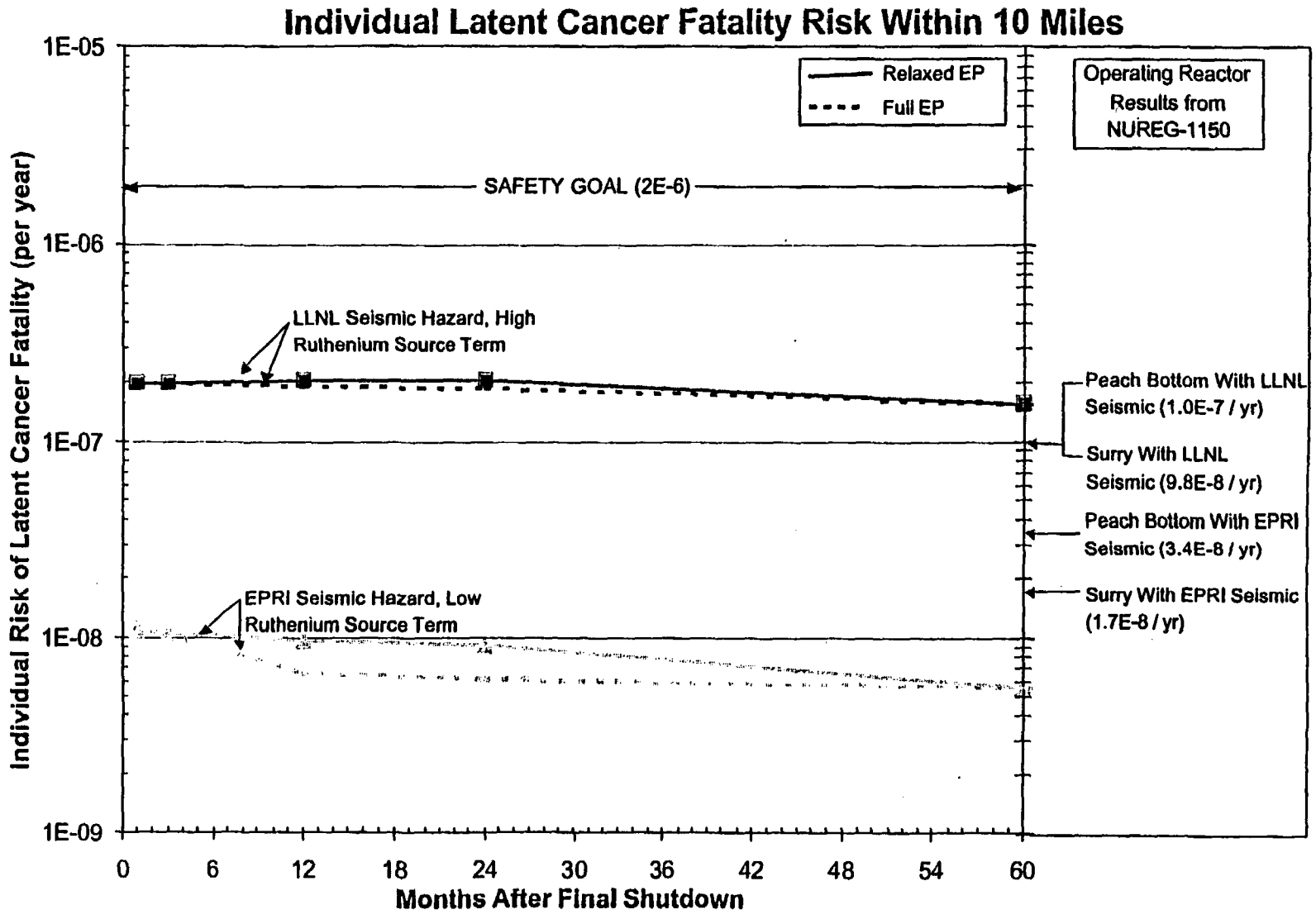
Individual Latent Cancer Fatality Risk Within 10 Miles
Relaxed EP
Full EP
Operating Reactor Results from NUREG-1150
SAFETY GOAL (2E-6)
LLNL Seismic Hazard, High Ruthenium Source Term
EPRI Seismic Hazard, Low Ruthenium Source Term
Peach Bottom With LLNL Seismic (1.0E-7 / yr)
Surry With LLNL Seismic (9.8E-8 / yr)
Peach Bottom With EPRI Seismic (3.4E-8 / yr)
Surry With EPRI Seismic (1.7E-8 / yr)
Individual Risk of Latent Cancer Fatality (per year)
1E-05
1E-06
1E-07
1E-08
1E-09
0
6
12
18
24
30
36
42
48
54
60
Months After Final Shutdown

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1.0 INTRODUCTION

Decommissioning plants have requested exemptions to certain regulations as a result of their permanently defueled condition. Although the current Part 50 regulatory requirements (developed for operating reactors) ensure safety at the decommissioning facility, some of these requirements may be excessive and not substantially contribute to public safety. Areas where regulatory relief has been requested in the past include exemptions from offsite emergency planning (EP), insurance, and safeguards requirements. Requests for consideration of changes in regulatory requirements are appropriate since the traditional accident sequences that dominate operating reactor risk are no longer applicable. For a defueled reactor in decommissioning status, public risk is predominantly from potential accidents involving spent fuel. Spent fuel can be stored in the spent fuel pool (SFP) for considerable periods of time, as remaining portions of the plant continue through decommissioning and disassembly. To date, exemptions have been requested and granted on a plant-specific basis. This has resulted in some inconsistency in the scope of evaluations and the acceptance criteria applied in processing the exemption requests.

To improve regulatory consistency and predictability, the NRC undertook this effort to improve the regulatory framework applicable to decommissioning plants. This framework utilized risk-informed approaches to identify the design and operational features necessary to ensure that risks to the public from these shutdown facilities are sufficiently small. This framework forms a technical foundation to be used as one input to developing regulatory changes, as well as a part of the basis for requesting and approving exemption requests until rulemaking is completed.

In support of this objective, the NRC staff has completed an assessment of SFP risks. This assessment utilized probabilistic risk assessment (PRA) methods and was developed from analytical studies in the areas of thermal hydraulics, reactivity, systems analysis, human reliability analysis, seismic and structural analysis, external hazards assessment, and offsite radiological consequences. The focus of the risk assessment was to identify potential severe accident scenarios at decommissioning plants and to estimate the likelihood and consequences of these scenarios. The staff also examined the offsite EP for decommissioning plants using an analysis strategy consistent with the principles of Regulatory Guide (RG) 1.174.

Preliminary versions of this study were issued for public comment and technical review in June 1999 and February 2000. Comments received from stakeholders and other technical reviewers have been considered in preparing this assessment. Quality assessment of the staff's preliminary analysis has been aided by a small panel of human reliability analysis (HRA) experts who evaluated the human performance analysis assumptions, methods and modeling. A broad quality review was carried out at the Idaho National Engineering and Environmental Laboratory (INEEL).

The study provides insights for the design and operation of SFP cooling and inventory makeup systems and practices and procedures necessary to ensure high levels of operator performance during off-normal conditions. The study concludes that, with the fulfillment of industry commitments and satisfaction of a number of important staff assumptions, the risks from SFPs can be sufficiently low to evaluate exemptions involving small changes to risk parameters and to contribute to the basis for related rulemaking.

As a measure of whether the risks from SFPs at decommissioning plants were sufficiently low to allow small changes to risk parameters, the concept of a pool performance guideline (PPG) was presented in the February 2000 study based on the principles of RG 1.174. In the study, the staff stated that consequences of an SFP fire are sufficiently severe that the RG 1.174 large early release frequency baseline of $1x10^{-5}$ per reactor year is an appropriate frequency guideline for a decommissioning plant SFP risk and a useful measure in combination with other factors such as accident progression timing, for assessing features, systems, and operator performance for a spent fuel pool in a decommissioning plant. Like the February 2000 study, this study uses the PPG of $1x10^{-5}$ per reactor year as the baseline frequency for a zirconium fire in the SFP.

The study is divided into three main parts. The first (Section 2) is a summary of the thermal-hydraulic analysis performed for SFPs at decommissioning plants. The second (Section 3) discusses how the principles of risk informed regulation are addressed by proposed changes. The third (Section 4) discusses the implications of the study for decommissioning regulatory requirements.

2.0 THERMAL-HYDRAULIC ANALYSES

Analyses were performed to evaluate the thermal-hydraulic characteristics of spent fuel stored in the spent fuel pools (SFPs) of decommissioning plants and determine the time available for plant operators to take actions to prevent a zirconium fire. These are discussed in Appendix 1A. The focus was the time available before fuel uncovery and the time available before the zirconium ignites after fuel uncovery. These times were utilized in performing the risk assessment discussed in Section 3.

To establish the times available before fuel uncovery, calculations were performed to determine the time to heat the SFP coolant to a point of boiling and then boil the coolant down to 3 feet above the top of the fuel. As can be seen in Table 2.1 below, the time available to take actions before any fuel uncovery is 100 hours or more for an SFP in which pressurized-water reactor (PWR) fuel has decayed at least 60 days.

Table 2.1 Time to Heatup and Boiloff SFP Inventory Down to 3 Feet Above Top of Fuel (60 GWD/MTU)

DECAY TIME	PWR	BWR
60 days	100 hours (>4 days)	145 hours (>6 days)
1 year	195 hours (>8 days)	253 hours (>10 days)
2 years	272 hours (>11 days)	337 hours (>14 days)
5 years	400 hours (>16 days)	459 hours (>19 days)
10 years	476 hours (>19 days)	532 hours (>22 days)

The analyses in Appendix 1A determined that the amount of time available (after complete fuel uncovery) before a zirconium fire depends on various factors, including decay heat rate, fuel burnup, fuel storage configuration, building ventilation rates and air flow paths, and fuel cladding oxidation rates. While the February 2000 study indicated that for the cases analyzed a required decay time of 5 years would preclude a zirconium fire, the revised analyses show that it is not feasible, without numerous constraints, to define a generic decay heat level (and therefore decay time) beyond which a zirconium fire is not physically possible. Heat removal is very sensitive to these constraints, and two of these constraints, fuel assembly geometry and spent fuel pool rack configuration, are plant specific. Both are also subject to unpredictable changes as a result of the severe seismic, cask drop, and possibly other dynamic events which could rapidly drain the pool. Therefore, since the decay heat source remains nonnegligible for many years and since configurations that ensure sufficient air flow[2] for cooling cannot be assured, a zirconium

[2]Although a reduced air flow condition could reduce the oxygen levels to a point where a fire would not be possible, there is sufficient uncertainty in the available data as to when this level would be reached and if it could be maintained. It is not possible to predict when a zirconium fire would not occur because of a lack of oxygen. Blockage of the air flow around the fuel could be

fire cannot be precluded, although the likelihood may be reduced by accident management measures.

Figure 2.1 plots the heatup time air-cooled PWR and BWR fuel take to heat up from 30 °C to 900 °C versus time since reactor shutdown. The figure shows that after 4 years, PWR fuel could reach the point of fission product release in about 24 hours. Figure 2.2 shows the timing of the event by comparing the air-cooled calculations to an adiabatic heatup calculation for PWR fuel with a burnup of 60 GWD/MTU. The figure indicates an unrealistic result that until 2 years have passed the air-cooled heatup rates are faster than the adiabatic heatup rates. This is because the air-cooled case includes heat addition from oxidation while the adiabatic case does not. In the early years after shutdown, the additional heat source from oxidation at higher temperatures is high enough to offset any benefit from air cooling. This result is discussed further in Appendix 1A. The results using obstructed airflow (adiabatic heatup) show that at 5 years after shutdown, the release of fission products may occur approximately 24 hours after the accident.

In summary, 60 days after reactor shutdown for boildown type events, there is considerable time (>100 hours) to take action to preclude a fission product release or zirconium fire before uncovering the top of the fuel. However, if the fuel is uncovered, heatup to the zirconium ignition temperature during the first years after shutdown would take less than 10 hours even with unobstructed air flow. After 5 years, the heatup would take at least 24 hours even with obstructed air flow cases. Therefore, a zirconium fire would still be possible after 5 years for cases involving obstructed air flow and unsuccessful accident management measures. These results and how they affect SFP risk and decommissioning regulations are discussed in Sections 3 and 4 of this study.

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caused by collapsed structures and/or a partial draindown of the SFP coolant or by reconfiguration of the fuel assemblies during a seismic event or heavy load drop. A loss of SFP building ventilation could also preclude or inhibit effective cooling. As discussed in Appendix 1A, air flow blockage without any recovery actions could result in a near-adiabatic fuel heatup and a zirconium fire even after 5 years.

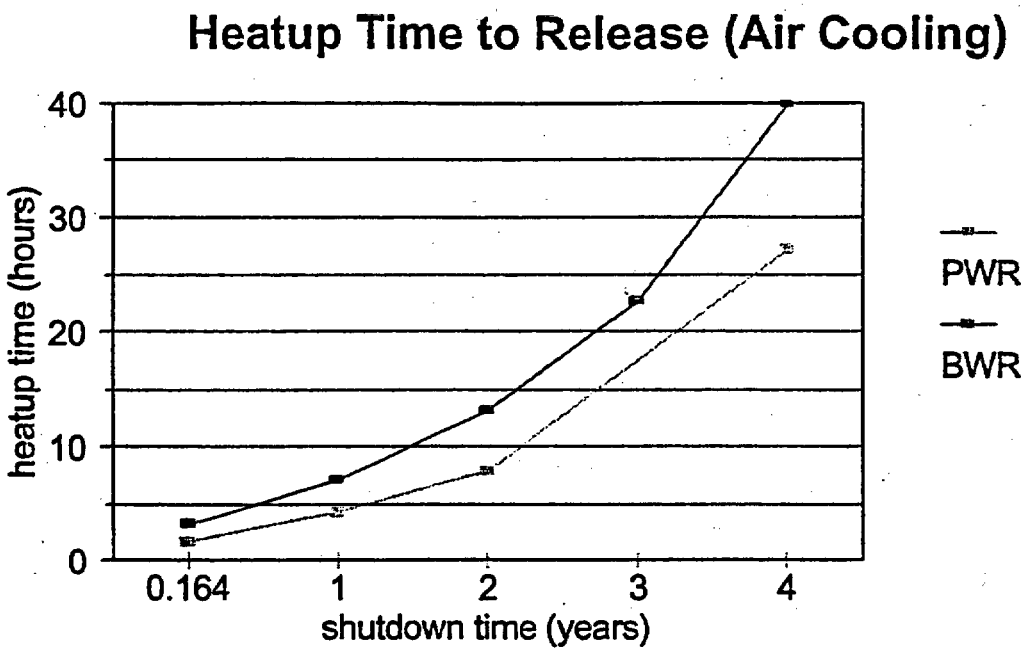

Figure 2.1 Heatup Time From 30 °C to 900 °C

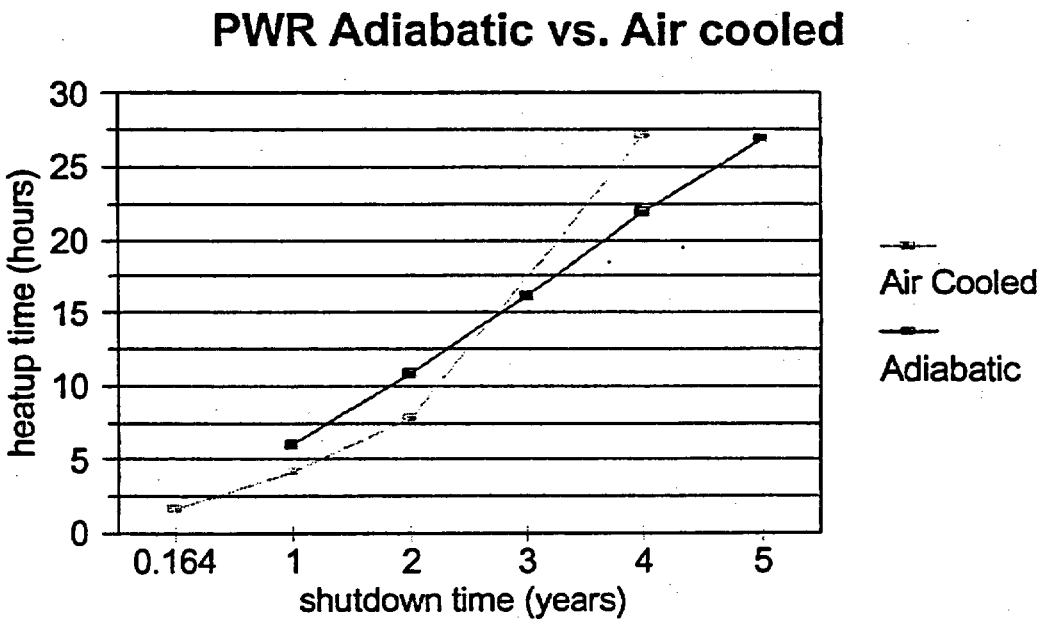

Figure 2.2 PWR Heatup Times for Air Cooling and Adiabatic Heatup

3.0 RISK ASSESSMENT OF SPENT FUEL POOLS AT DECOMMISSIONING PLANTS

The scenarios leading to significant offsite consequences at a decommissioning plant are different than at an operating plant. Once fuel is permanently removed from the reactor vessel, the primary public risk in a decommissioning facility is associated with the spent fuel pool (SFP). The spent fuel assemblies are retained in the SFP and submerged in water to cool the remaining decay heat and to shield the radioactive assemblies. The most severe accidents postulated for SFPs are associated with the loss of water from the pool.

Depending on the time since reactor shutdown, fuel burnup, and fuel rack configuration, there may be sufficient decay heat for the fuel clad to heat up, swell, and burst after a loss of pool water. The breach in the clad releases of radioactive gases present in the gap between the fuel and clad. This is called "a gap release" (see Appendix 1B). If the fuel continues to heat up, the zirconium clad will reach the point of rapid oxidation in air. This reaction of zirconium and air, or zirconium and steam is exothermic (i.e., produces heat). The energy released from the reaction, combined with the fuel's decay energy, can cause the reaction to become self-sustaining and ignite the zirconium. The increase in heat from the oxidation reaction can also raise the temperature in adjacent fuel assemblies and propagate the oxidation reaction. The zirconium fire would result in a significant release of the spent fuel fission products which would be dispersed from the reactor site in the thermal plume from the zirconium fire. Consequence assessments (Appendix 4) have shown that a zirconium fire could have significant latent health effects and resulted in a number of early fatalities. Gap releases from fuel from a reactor that has been shutdown more than a few months involve smaller quantities of radionuclides and, in the absence of a zirconium fire, would only be of concern onsite.

The staff conducted its risk evaluation to estimate the likelihood of accident scenarios that could result in loss of pool water and fuel heatup to the point of rapid oxidation. In addition to developing an assessment of the level of risk associated with SFPs at decommissioning plants, the staff's objective was to identify potential vulnerabilities and design and operational characteristics that would minimize these vulnerabilities. Finally, the staff assessed the effect of offsite emergency planning (i.e., evacuation) at selected sites using various risk metrics and the Commission's Safety Goals.

In support of the risk evaluation, the staff conducted a thermal-hydraulic assessment of the SFP for various scenarios involving loss of pool cooling and loss of inventory. These calculations provided information on heatup and boiloff rates for the pool and on heatup rates for the uncovered fuel assemblies and time to initiation of a zirconium fire (see Table 2.1 and Figures 2.1 and 2.2). The results of these calculations provided fundamental information on the timing of accident sequences, insights on the time available to recover from events, and time available to initiate offsite measures. This information was used in the risk assessment to support the human reliability analysis of the likelihood of refilling the SFP or cooling the fuel before a zirconium fire occurs.

For these calculations, the end state assumed for the accident sequences was the state at which the water level reached 3 feet from the top of the spent fuel. This simplification was used because of the lack of data and difficulty in modeling complex heat transfer mechanisms and chemical reactions in the fuel assemblies that are slowly being uncovered. As a result, the time available for fuel handler recovery from SFP events before initiation of a zirconium fire is

underestimated. However, since recoverable events such as small loss of inventory or loss of power or pool cooling evolve very slowly, many days are generally available for recovery whether the end point of the analysis is uncovery of the top of the fuel or complete fuel uncovery. The extra time available (estimated to be in the tens of hours) as the water boils off would not impact the very high probabilities of fuel handler recovery from these events, given the industry decommissioning commitments (IDCs) and additional staff decommissioning assumptions (SDAs) discussed in Sections 3.2 through 4.[3] A summary of the thermal-hydraulic assessment is provided in Appendix 1A.

3.1 Basis and Findings of SFP Risk Assessment

To gather information on SFP design and operational characteristics for the preliminary risk assessment for the June 1999 draft study, the staff visited four decommissioning plants to ascertain what would be an appropriate model for decommissioning SFPs. The site visits confirmed that the as-operated SFP cooling systems were different from those in operation when the plants were in power operation. The operating plant pool cooling and makeup systems generally have been removed and replaced with portable, skid-mounted pumps and heat exchangers. In some cases there are redundant pumps. In most cases, physical separation, barrier protection, and emergency onsite power sources are no longer maintained. Modeling information for the PRA analysis was gathered from system walkdowns and discussions with the decommissioning plant staff. Since limited information was collected for the preliminary assessment on procedural and recovery activities and on the minimum configuration for a decommissioning plant, a number of assumptions and bounding conditions were in the June 1999 study. The preliminary results have been refined in this assessment, thanks to more detailed information from industry on SFP design and operating characteristics for a decommissioning plant and a number of IDCs that contribute to achieving low risk findings from SFP incidents. The revised results also reflect improvements in the PRA model since publication of the June 1999 and February 2000 studies.

The staff identified nine initiating event categories to investigate as part of the quantitative assessment on SFP risk:

1. Loss of offsite power from plant centered and grid-related events
2. Loss of offsite power from events initiated by severe weather
3. Internal fire
4. Loss of pool cooling
5. Loss of coolant inventory
6. Seismic event
7. Cask drop
8. Aircraft impact
9. Tornado missile

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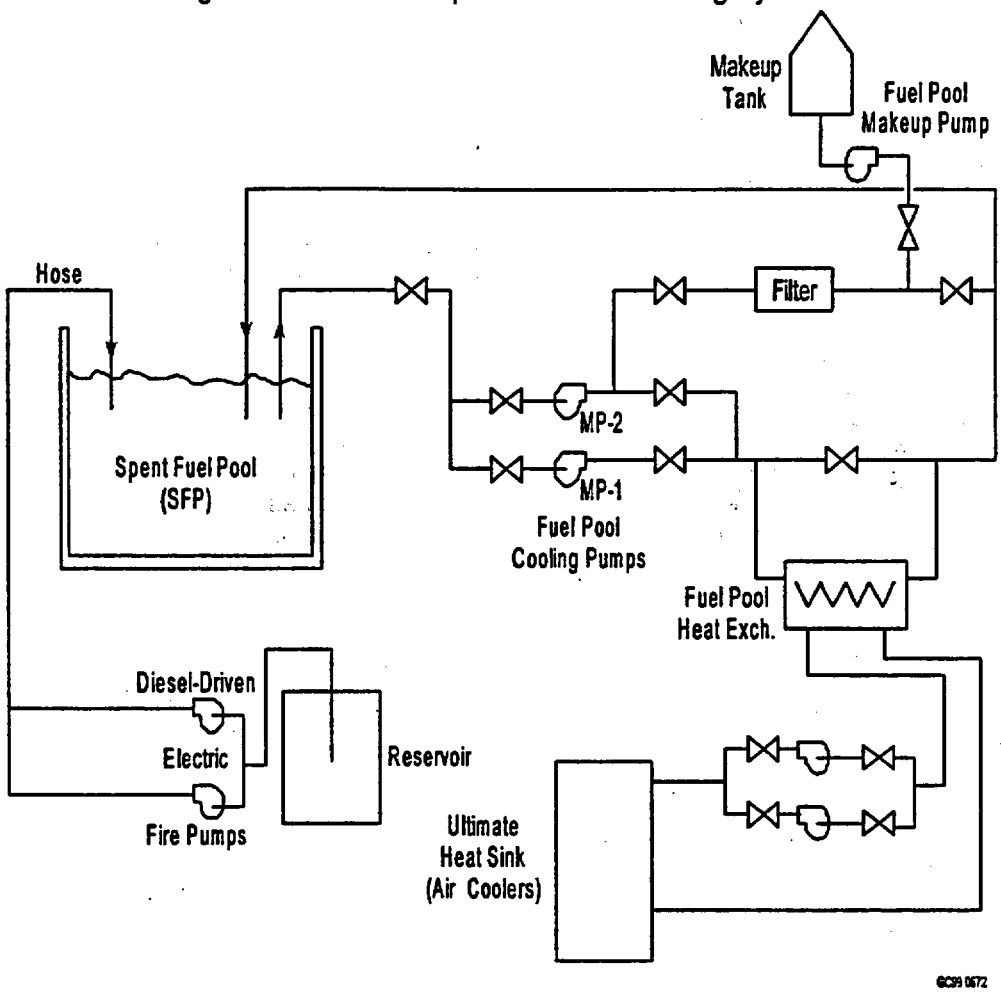
[3]The staff notes that the assumption that no recovery occurs once the water level reaches 3 feet above the fuel tends to obscure the distinction between two major types of accidents: slow boildown or draindown events and rapid draindown events. In both types of events, cooling would most likely be not by air but by water or steam. Also obscured is the effect of partial draindown events on event timing (addressed in Section 2).

In addition, a qualitative risk perspective was developed for inadvertent criticality in the SFP (see Section 3.6). The risk model, as developed by the staff and revised after a quality review by Idaho National Engineering & Environmental Laboratory (INEEL), is provided in Appendix 2A. Appendix 2A also includes the modeling details for the heavy load drop, aircraft impact, seismic, and tornado missile assessments. Input and comments from stakeholders were also utilized in updating the June 1999 and February 2000 risk models.

3.2 Characteristics of SFP Design and Operations for a Decommissioning Plant

Based on information gathered from the site visits and interactions with NEI and other stakeholders, the staff modeled the spent fuel pool cooling (SFPC) system (see Figure 3.1) as being located in the SFP area and consisting of motor-driven pumps, a heat exchanger, an ultimate heat sink, a makeup tank, a filtration system, and isolation valves. Coolant is drawn from the SFP by one of the two pumps, passed through the heat exchanger, and returned to the pool. One of the two pumps on the secondary side of the heat exchanger rejects the heat to the ultimate heat sink. A small amount of water is diverted to the filtration process and is returned into the discharge line. A manually operated makeup system (with a limited volumetric flow rate) supplements the small losses due to evaporation. During a prolonged loss of the SFPC system or a loss of inventory, inventory can be made up using the firewater system, if needed. Two firewater pumps, one motor-driven (electric) and one diesel-driven, provide firewater in the SFP area. There is a firewater hose station in the SFP area. The firewater pumps are in a separate structure.

Figure 3.1 Assumed Spent Fuel Pool Cooling System



Based upon information obtained during the site visits and discussions with decommissioning plant personnel during those visits, the staff also made the following assumptions that are believed to be representative of a typical decommissioning facility:

- The SFP cooling design, including instrumentation, is at least as capable as that assumed in the risk assessment. Licensees have at least one motor-driven and one diesel-driven fire pump capable of delivering inventory to the SFP (SDA #1, Table 4.2-2).
- The makeup capacity (with respect to volumetric flow) is assumed to be as follows:

Makeup pump:	20 – 30 gpm
Firewater pump:	100 – 200 gpm
Fire engine:	100 – 250 gpm (100 gpm, for hose: 1½-in., 250 gpm for 21/2-in. hose)

- For the larger loss-of-coolant-inventory accidents, water addition through the makeup pumps does not successfully mitigate the loss of the inventory event unless the location of inventory loss is isolated.

- The SFP fuel handlers perform walkdowns of the SFP area once per shift (8- to 12-hour shifts). A different crew member works the next shift. The SFP water is clear and the pool level is observable via a measuring stick in the pool to alert fuel handlers to level changes.

- Plants do not have drain paths in their SFPs that could lower the pool level (by draining, suction, or pumping) more that 15 feet below the normal pool operating level, and licensees must initiate recovery using offsite sources.

Based upon the results of the June 1999 preliminary risk analysis and the associated sensitivity cases, it became clear that many of the risk sequences were quite sensitive to the performance of the SFP operating staff in identifying and responding to off-normal conditions. This is because the remaining systems of the SFP are relatively simple, with manual rather than automatic initiation of backups or realignments. Therefore, in scenarios such as loss of cooling or inventory loss, the fuel handler's responses to diagnose the failures and bring any available resources (public or private) to bear is fundamental for ensuring that the fuel assemblies remain cooled and a zirconium fire is prevented.

As part of its technical evaluations, the staff assembled a small panel of experts[4] to identify the attributes necessary to achieving very high levels of human reliability for responding to potential accident scenarios in a decommissioning plant SFP. (These attributes and the human reliability analysis (HRA) methodology used are discussed in Section 3.2 of Appendix 2A.)

Upon considering the sensitivities identified in the staff's preliminary study and to reflect actual operating practices at decommissioning facilities, the nuclear industry, through NEI, made important commitments, which are reflected in the staff's updated risk assessment.

Industry Decommissioning Commitments (IDCs)

IDC #1 Cask drop analyses will be performed or single failure-proof cranes will be in use for handling of heavy loads (i.e., phase II of NUREG-0612 will be implemented).

IDC #2 Procedures and training of personnel will be in place to ensure that onsite and offsite resources can be brought to bear during an event.

IDC #3 Procedures will be in place to establish communication between onsite and offsite organizations during severe weather and seismic events.

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[4]Gareth Parry, U.S. NRC; Harold Blackman, INEEL; and Dennis Bley, Buttonwood Consulting.

IDC #4 An offsite resource plan will be developed which will include access to portable pumps and emergency power to supplement onsite resources. The plan would principally identify organizations or suppliers where offsite resources could be obtained in a timely manner.

IDC #5 Spent fuel pool instrumentation will include readouts and alarms in the control room (or where personnel are stationed) for spent fuel pool temperature, water level, and area radiation levels.

IDC #6 Spent fuel pool seals that could cause leakage leading to fuel uncovery in the event of seal failure shall be self limiting to leakage or otherwise engineered so that drainage cannot occur.

IDC #7 Procedures or administrative controls to reduce the likelihood of rapid draindown events will include (1) prohibitions on the use of pumps that lack adequate siphon protection or (2) controls for pump suction and discharge points. The functionality of anti-siphon devices will be periodically verified.

IDC #8 An onsite restoration plan will be in place to provide repair of the spent fuel pool cooling systems or to provide access for makeup water to the spent fuel pool. The plan will provide for remote alignment of the makeup source to the spent fuel pool without requiring entry to the refuel floor.

IDC #9 Procedures will be in place to control spent fuel pool operations that have the potential to rapidly decrease spent fuel pool inventory. These administrative controls may require additional operations or management review, management physical presence for designated operations or administrative limitations such as restrictions on heavy load movements.

IDC #10 Routine testing of the alternative fuel pool makeup system components will be performed and administrative controls for equipment out of service will be implemented to provide added assurance that the components would be available, if needed.

Additional important operational and design assumptions made by the staff in the risk estimates developed in this study are designated as SDAs and are discussed in later sections of this study.

3.3 Estimated Frequencies of Spent Fuel Uncovery and Assumptions That Influence the Results

Based upon the above design and operational features, IDCs, technical comments from stakeholders, and the input from the INEEL technical review, the staff's SFP risk model was updated. The updates have improved the estimated frequency calculations, but have not changed the need for the industry commitments or staff decommissioning assumptions. Absolute values of some sequences have decreased, but the overall insights from the risk assessment remain the same.

3.3.1 Internal and External Initiator Frequency of Spent Fuel Pool Uncovery

The results for the initiators that were assessed quantitatively are shown in Table 3.1. This table gives the fuel uncovery frequency for each accident initiator. The frequencies are point estimates because point estimates were used for the input parameters. For the most part, these input parameter values are the mean values of the probability distributions that would be used in a calculation to propagate parameter uncertainty. Because the systems are very simple and needs little support, the point estimates therefore reasonably correlate to the mean values that would be obtained from a full propagation of parameter uncertainty. Due to the large margin between the loss of cooling and inventory sequence frequencies and the pool performance guideline, this propagation was judged to be unnecessary (see Section 5.1 of Appendix 2A for further discussion of uncertainties).

Both the EPRI and LLNL hazard estimates at reactor sites were developed as best estimates and are considered valid by the NRC. Furthermore, because both sets of curves are based upon expert opinion and extrapolation in the range of interest, there is no technical basis for excluding consideration of either set of estimates. The mean frequency shown does not consider Western U.S. sites (e.g., Diablo Canyon, San Onofre, and WNP-2).

The results in Table 3.1 show the estimated frequency of a zirconium fire of fuel that has a decay time of 1 year. In characterizing the risk of seismically-induced SFP accidents for the population of sites, the staff has displayed results based on both the LLNL and EPRI hazard estimates, and has used an accident frequency corresponding to the mean value for the respective distributions, i.e., a frequency of $2x10^{-6}$ per year to reflect the use of LLNL hazard estimates and a frequency of $2x10^{-7}$ per year to reflect use of the EPRI hazard estimates. Use of the mean value facilitates comparisons with the Commission's Safety Goals and QHOs. Fire frequencies for all initiators range from about $6x10^{-7}$ per year to about $2X10^{-6}$ per year (depending on the seismic hazard estimates used), with the dominant contribution being from a severe seismic event. Plant-specific frequency estimates in some cases could be as much as an order of magnitude higher or lower because of the seismic hazard at the plant site. The mean value bounds about 70 percent of the sites for either the LLNL or the EPRI cases. A more detailed characterization of the seismic risk is given in Section 3.5.1 and Appendix 2B. The frequency of a zirconium fire is dominated by seismic events when the seismic hazard frequency is based on the LLNL estimate. Cask drop and boildown sequences become important contributors when seismic hazard frequency is based on the EPRI estimate. As a result, even though the seismic event frequency based on the EPRI estimate is an order of magnitude lower than the LLNL estimate, only a factor of four reduction in total frequency is realized with the use of the EPRI estimate since the nonseismic sequences become more important. In Section 3.4.7 the staff discusses the expected fuel uncovery frequencies for fuel that has decayed a few months, 2 years, 5 years, and 10 years.

In conjunction with the frequency of the uncovery of the spent fuel, it is important to know the time it takes the fuel to heat up once it has been uncovered fully or partially. Figures 2.1 and 2.2 in Section 2 show the time needed with and without air circulation to heat up the fuel from 30 °C to 900 °C (the temperature at which zirconium oxidation is postulated to become runaway oxidation and at which fission products are expected to be expelled from the fuel and cladding).

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The staff realizes that the volumetric rate of air flow that a fuel bundle receives during a loss of cooling event significantly influences the heatup of the bundle. To achieve sufficient long-term air cooling of uncovered spent fuel, two conditions must be met: (1) an air flow path through the bundles must exist, and (2) sufficient SFP building ventilation flow must be provided. The presence of more than about 1 foot of water in the SFP, as in a seismically induced SFP failure or the late states of a boildown sequence, would effectively block the air flow path. Seismically induced collapse of the SFP building into the SFP could have a similar effect. Loss of building ventilation would tend to increase fuel heatup rates and maximum fuel temperatures, as described in Appendix 1A.

Based on engineering judgment, we have partitioned the frequency of each sequence into two parts: where the bundles in the spent fuel pool area receive two building volumes of air per hour (high air flow) and where the bundles receives little or no air flow (low air flow). Table 3.2 provides this partition.

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Table 3.1 Spent Fuel Pool Cooling Risk Analysis — Frequency of Fuel Uncovery (per year)

INITIATING EVENT	Frequency of Fuel Uncovery (EPRI hazard)	Frequency of Fuel Uncovery (LLNL hazard)
Seismic event [5]	$2x10^{-07}$	$2x10^{-06}$
Cask drop [6]	$2.0x10^{-07}$	same
Loss of offsite power[7] initiated by severe weather	$1.1x10^{-07}$	same
Loss of offsite power from plant centered and grid-related events	$2.9x10^{-08}$	same
Internal fire	$2.3x10^{-08}$	same
Loss of pool cooling	$1.4x10^{-08}$	same
Loss of coolant inventory	$3.0x10^{-09}$	same
Aircraft impact	$2.9x10^{-09}$	same
Tornado Missile	$<1.0x10^{-09}$	same
Total[8]	$5.8x10^{-07}$	$2.4x10^{-06}$

[5]This value is the mean of the failure probabilities for Central and Eastern SFPs that satisfy the seismic checklist and includes seismically induced catastrophic failure of the pool (which dominates the results) and a small contribution from seismically induced failure of pool support systems.

[6]For a single-failure-proof system without a load drop analysis. The staff assumed that facilities that chose the option in NUREG-0612 have a non-single-failure-proof system and implemented their load drop analysis including taking mitigative actions to assure a high confidence that the risk of catastrophic failure was less than or equivalent to that of a single-failure-proof system.

[7]The estimate is based upon the time available for human response when the fuel has decayed 1 year. After only a few months of decay, these estimates are not expected to increase significantly. Furthermore, for longer periods of decay, no significant change in the estimated frequency is expected because the fuel handler success rates are already so high after 1 year of decay.

[8]Consistent with PRA limitations and practice, contributions to risk from safeguards events are not included in these frequency estimates. EP might also provide dose savings in such events.

Table 3.2 Spent Fuel Pool Cooling Risk Analysis — Frequency Partition for Air Flow

SEQUENCES	% HIGH AIR FLOW	% LOW AIR FLOW (ADIABATIC)
Seismic	30%	70%
Heavy load drop	50%	50%
Loss of offsite power, severe weather	90%	10%

In Table 3.2 for seismic sequences, we have assumed that 30 percent of the time the building will turn over two building volumes of air per hour (high air flow) and 70 percent of the time the individual fuel bundle of concern will receive little or no air cooling. These percentages are based on discussions with staff structural engineers, who believe that, at accelerations in excess of 1.2 g spectral acceleration (which is greater than three times the safe shutdown earthquake (SSE) for many reactor sites east of the Rocky Mountains), there is a high likelihood of building damage that blocks air flow. For heavy load drop sequences, the staff assumed a 50 percent partition for the high air flow case. This is based on considering both damage to fuel bundles due to a heavy load drop that renders bundles uncoolable and the alternative possibility that the drop damages the building structure in a way that blocks some spent fuel bundles. For loss of offsite power events caused by severe weather, the staff assumed a 90 percent partition for the high airflow case. This is based on a staff assumption that openings in the SFP building (e.g., doors and roof hatches) are large enough that, if forced circulation is lost, natural circulation cooling will provide at least two building volume of air per hour to the SFP. This assumption may need to be confirmed on a plant-specific basis. The staff did not partition the rest of the sequences in Table 3.2, since their absolute value and contribution to the overall zirconium fire frequency are so low.

The frequency partitioning shows that a large portion of the seismic and heavy load drop sequences have low air flow. This partitioning did not consider the possibility that the air flow path is blocked by residual water in the SFP. When the potential for flow blockage by residual water is considered, an even greater portion of the events would result in no (or low) air flow. Fuel heatup calculations in Section 2 show that for the first several years after shutdown, the fuel heatup time (e.g., time to reach 900 °C) for the adiabatic and air-cooled cases is comparable. Thus, the effects of partitioning are negligible for this period. Because SFP and SFP building fragilities and failure modes are plant-specific, and the heatup time for the adiabatic and air-cooled cases differ only slightly, the staff did not consider the partitioning in estimating the frequency of SFP fires. Whether or not a spent fuel bundle receives high air flow or low air flow fuel uncovery does not change our insights into the risk associated with operation of SFPs.

3.3.2 Important Assumptions

As discussed in more detail in Appendix 2A, the results of the risk analysis depend on assumptions about the design and operational characteristics of the SFP facility. The following inputs can significantly influence the results:

- The modeled system configuration is described in Section 3.2. The assumed availability of a diesel-driven fire pump is an important factor in the conclusion that fuel uncovery frequency is low for the loss of offsite power initiating events and the internal fire initiating event. The assumption of the availability of a redundant fuel pool cooling pump is not as important since the modeling of the recovery of the failed system includes repair of the failed pump as well as the startup of the redundant pump. Finally, multiple sources of makeup water are assumed for the fire pumps. This lessens the possible dependencies between initiating events (e.g., severe weather, high winds, or earthquakes) and the availability of makeup water supply (e.g., the fragility of the fire water supply tank).

- Plants have no drain paths in their SFPs that could lower the pool level (by draining, suction, or pumping) more that 15 feet below the normal pool operating level, and licensees must initiate recovery using offsite sources.

- Openings in the SFP building (e.g., doors and roof hatches) are large enough that, if forced circulation is lost, natural circulation cooling will provide at least two building volumes of air per hour to the SFP. Procedures exist to implement natural circulation.

- Credit is taken for the industry/NEI commitments as described in Section 3.2. Without this credit, the risk is estimated to be more than an order of magnitude higher. Specifically —

  - S IDC #1 is credited for lowering the risk from cask drop accidents.

  - S IDCs #2, 3, 4, and 8 are credited for the high probability of recovery from loss of cooling scenarios (including events initiated by loss of power or fire) and loss of inventory scenarios. To take full credit for these commitments, additional assumptions have been made about how these commitments will be implemented. Procedures and training are assume to give explicit guidance on the capability of the fuel pool makeup system and on when it becomes essential to supplement with alternate higher volume sources. Procedures and training are assumed to give sufficiently clear guidance on early preparation for using the alternate makeup sources. Walkdowns are assumed once per shift and the fuel handlers are assumed to document their observations in a log. The last assumption compensates for potential failures of the instrumentation monitoring the status of the pool.

  - S IDC #5 is credited for the high probability of early identification and diagnosis (from the control room) of loss of cooling or loss of inventory.

  - S IDCs #6, 7, and 9 are credited with lowering the initiating event frequency for the loss of inventory event from historical levels. In addition, these commitments are used to justify the assumption that a large noncatastrophic leak rate is limited to approximately 60 gpm and the assumption that the leak is self-limiting after a drop in level of 15 feet. These

assumptions may be nonconservative on a plant-specific basis depending on SFP configuration and commitments for configuration control.

S IDC #10 is credited for the equipment availabilities and relibilities used in the analysis. In addition, if there are administrative procedures to control the out-of-service duration for the diesel fire pump, the relatively high unavailability for this pump (0.18) could be lowered.

- Initiating event frequencies for loss of cooling, loss of inventory, and loss of offsite power are based on generic data. In addition, the probability of power recovery is also based on generic information. Site-specific differences will proportionately affect the risk from these initiating events.

The various initiating event categories are discussed below. The staff's qualitative risk insights on the potential for SFP criticality are discussed in Section 3.6.

3.4 Internal Event Scenarios Leading to Fuel Uncovery

This section describes the events associated with internal event initiators. More details are given in Appendix 2A.

3.4.1 Loss of Cooling

The loss of cooling initiating event may be caused by the failure of pumps or valves, by piping failures, by an ineffective heat sink (e.g., loss of heat exchangers), or by a local loss of power (e.g., electrical connections). Although it may not be directly applicable because of design differences in decommissioning plants, operational data from NUREG-1275, Volume 12 (Ref. 3), shows that the frequency of loss of SFP cooling events in which temperature increases more than 20 °F is on the order of two to three events per 1000 reactor years. The data also shows that was the loss of cooling lasted less than 1 hour. Only three events exceeded 24 hours: the longest was 32 hours. In four events the temperature increase exceeded 20 °F, the largest increase being 50 °F.

The calculated fuel uncovery frequency for this initiating event is 1.4×10^{-8} per year. Indications of a loss of pool cooling available to fuel handlers include control room alarms and indicators, local temperature measurements, increasing area temperature and humidity, and low pool water level from boiloff. For a fuel uncovery, the plant fuel handlers must fail to recover the cooling system (either fail to notice the loss of cooling indications or fail to repair or restore the cooling system). In addition, the fuel handlers must fail to provide makeup cooling using other onsite sources (e.g., fire pumps) or offsite sources (e.g., a fire brigade). A long time is available for these recovery actions. In the case of 1-year-old fuel (i.e., fuel that was in the reactor when it was shutdown 1 year ago), approximately 195 hours is available for a PWR and 253 hours for a BWR before the water level drops to within 3 feet of the spent fuel. If the fuel most recently offloaded is only 2 months out of the reactor, the time available is still long (100–150 hours), and the likelihood of fuel handler success is still very high. These heatup and boiloff times are about double those reported by the staff before a correction in the staff's heat load assumptions. Because the uncovery frequency is already very low (on the order of 1 in 1 million per year) both absolutely and relative to other initiators, and because the quantification of human reliability

analysis values for such extended periods of recovery is beyond the state-of-the-art, the staff did not attempt to recalculate the expected uncovery frequency. For 2-year-old, 5-year-old, and 10-year-old fuel, much longer periods are available than at 1 year (see Table 2.1).

A careful and thorough adherence to IDCs #2, #5, #8, and #10 is crucial to establishing and maintaining the low frequency. In addition, however, the assumption that walkdowns are performed on a regular basis (once per shift) is important to compensate for potential failures of the instrumentation monitoring the status of the pool. The analysis has also assumed that the procedures and/or training give explicit guidance on the capability of the fuel pool makeup system and on when it becomes essential to supplement with alternative higher volume sources. The analysis also assumes that the procedures and training give sufficiently clear guidance on early preparation for using the alternative makeup sources.

There have been two recent events involving a loss of cooling at SFPs. The first, at Browns Ferry Unit 3 occurring in December 1998, involved a temperature increase of approximately 25 °F over a 2-day period. This event, caused by the short cycling of cooling water through a stuck-open check valve, was not detected by the control room indicators because of a design flaw in the indicators. In the second event, at the Duane Arnold Unit 1 in January 2000, the SFP temperature increased by 40 to 50 °F. The incident, which was undetected for approximately 2½ days, was caused by operator failure to restore the SFP cooling system heat sink after maintenance activities. The plant had no alarm for high fuel pool temperature, although there are temperature indicators in the control room. Since the conditional probability of fuel uncovery is low given a loss of cooling initiating event, the addition of these two recent events to the database will not affect the conclusion that the risk from these events is low. However, the recent events illustrate the importance of industry commitments, particularly IDC #5 which requires temperature instrumentation and alarms in the control room. In addition, the staff assumptions that walkdowns are performed on a regular basis (once per shift), with the fuel handler documenting the observations in a log, and the assumption that control room instrumentation that monitors SFP temperature and water level directly measures the parameters involved are important for keeping the risk low, since the walkdowns compensate for potential failures of the control room instrumentation and direct measurement precludes failures such as occurred at Browns Ferry.

Even with the above referenced industry commitments, the additional need for walkdowns to be performed at least once per shift and the specific need for direct indication of level and temperature had to be assumed in order to arrive at the low accident frequency calculated for this scenario. These additional assumptions are identified by the staff as staff decommissioning assumptions (SDAs) #2 and #3. SDA #2 assumes the existence of explicit procedures and fuel handler training to provide guidance on the capability and availability of inventory makeup sources and the time available to utilize these sources.

SDA #2 Walkdowns of SFP systems are performed at least once per shift by the fuel handlers. Procedures are in place to give the fuel handlers guidance on the capability and availability of onsite and offsite inventory makeup sources and on the time available to utilize these sources for various loss of cooling or inventory events.

SDA #3 Control room instrumentation that monitors SFP temperature and water level directly measures the parameters involved. Level instrumentation provides alarms for calling in offsite resources and for declaring a general emergency.

3.4.2 Loss of Coolant Inventory

This initiator includes loss of coolant inventory resulting from configuration control errors, siphoning, piping failures, and gate and seal failures. Operational data in NUREG-1275, Volume 12, shows that the frequency of loss of inventory events in which a level decrease of more than 1 foot occurred is less than one event per 100 reactor years. Most of these events are as a result of fuel handler errors and are recoverable. Many of the events are not applicable in a decommissioning facility.

NUREG-1275 shows that, except for one event that lasted 72 hours, no events lasted more than 24 hours. Eight events resulted in a level decrease of between 1 and 5 feet, and another two events resulted in an inventory loss of between 5 and 10 feet.

Using the information from NUREG-1275, it can be estimated that 6 percent of the loss of inventory events will be large enough and/or long enough to require that isolating the loss if the only system available for makeup is the SFP makeup system. For the other 94 percent of the cases, operation of the makeup pump is sufficient to prevent fuel uncovery.

The calculated fuel uncovery frequency for loss of inventory events is 3.0×10^{-9} per year. The uncovery frequency is low primarily due to the assumption that loss of inventory can drain the pool only so far. Once that level is reached, additional inventory loss must come from pool heatup and boiloff. Fuel uncovery occurs if plant fuel handlers fail to initiate inventory makeup either by use of onsite sources such as the fire pumps or offsite sources such as the local fire department. In the case of a large leak, isolation of the leak would also be necessary if the makeup pumps are used. The time available for fuel handler action is considerable, and even in the case of a large leak, it is estimated that 40 hours will be available. Fuel handlers are alerted to a loss of inventory condition by control room alarms and indicators, by the visibly dropping water level in the pool, by the accumulation of water in unexpected locations, and by local alarms (radiation alarms, building sump high level alarms, etc.).

As with the loss of pool cooling, the frequency of fuel uncovery is calculated to be very low. Again, a careful and thorough adherence to IDCs #2, #5, #8, and #10 is crucial to establishing the low frequency. In addition, the assumptions that walkdowns (see SDA #2 above) are performed on a regular basis (once per shift) and that instrumentation directly measures temperature and level are important to compensate for potential failures of the instrumentation monitoring the status of the pool. The assumption that the procedures and/or training give explicit guidance on the capability of the fuel pool makeup system lowers the expected probability of fuel handler human errors, and the assumption that fuel handlers will supplement SFP makeup at appropriate times from alternative higher volume sources lowers the estimated frequency of failure of the fuel handler to mitigate the loss of coolant inventory. IDCs #6, #7, and #9 are also credited with lowering the initiating event frequency.

Even with these industry commitments, the staff had to assume the drop in pool inventory due to loss of inventory events is limited in order to arrive at the low accident frequency calculated for this scenario. This additional assumption is identified by the staff as SDA #4.

SDA #4 The licensee has determined that the SFP has no drain paths that could lower the pool level (by draining, suction, or pumping) more than 15 feet below the normal pool operating level and that the licensee initiates recovery using offsite sources.

3.4.3 Loss of Offsite Power from Plant-Centered and Grid Related Events

A loss of offsite power from plant-centered events typically involves hardware failures, design deficiencies, human errors (in maintenance and switching), localized weather-induced faults (e.g., lightning), or combinations. Grid-related offsite power events are caused by problems in the offsite power grid. With the loss of offsite power (onsite power is lost too, since the staff assumes no diesel generator is available to pick up the necessary electrical loads), there is no effective way of removing heat from the SFP. If power is not restored in time, the pool will heatup and boiloff inventory until the fuel is uncovered. The diesel-driven fire pump is available to provide inventory makeup. If the diesel-driven pump fails, and if offsite power is not recovered promply, recovery using offsite fire engines is a possibility. Recovery times are the same as for loss of cooling (discussed in Section 3.4.1).

Even after recovering offsite power, the fuel handlers have to restart the fuel pool cooling pumps. Failure to do this or failure of the equipment to restart will necessitate other fuel handler recovery actions. Again, considerable time is available.

The calculated fuel uncovery frequency for this sequence of events is 2.9×10^{-8} per year. This frequency is very low and, as with loss of pool cooling and loss of inventory, is based on adherence to IDCs #2, #5, #8, and #10. In addition, regular plant walkdowns, clear and explicit procedures, fuel handler training (SDA #2), and the direct measurement of level and temperature in the SFP (SDA #3) are assumed as documented.

3.4.4 Loss of Offsite Power from Severe Weather Events

This event represents the loss of SFP cooling because of a loss of offsite power from severe weather-related events (hurricanes, snow and wind, ice, wind and salt, wind, and one tornado event). Because of the potential for severe localized damage, tornadoes are analyzed separately in Appendix 2E. The analysis is summarized in Section 3.5.3 of this study.

Until offsite power is recovered, the electrical pumps are unavailable and the diesel-driven fire pump is available only for makeup. Recovery of offsite power after severe weather events is assumed to be less probable than after grid-related and plant-centered events. In addition, it is more difficult for offsite help to reach the site.

The calculated fuel uncovery frequency for this event is 1.1×10^{-7} per year. As in the previous cases, this estimate was based on IDCs #2, #5, #8, #10 and on assumptions documented in SDA #2 and SDA #3. In addition, IDC #3, the commitment to have procedures in place for communications between onsite and offsite organizations during severe weather, is also

important in the analysis for increasing the likelihood that offsite organization can respond effectively.

3.4.5 Internal Fire

This event tree models the loss of SFP cooling caused by internal fires. The staff assumed that there is no automatic fire suppression system for the SFP cooling area. The fuel handler may initially attempt to manually suppress the fire if the fuel handler responds to the control room or local area alarms. If the fuel handler fails to respond to the alarm or is unsuccessful in extinguishing the fire within the first 20 minutes, the staff assumes that the SFP cooling system will be significantly damaged and cannot be repaired. Once the inventory level drops below the SFP cooling system suction level, the fuel handlers have about 85 hours to provide some sort of alternative makeup, either by using the site firewater system or by calling upon offsite resources. The staff assumes that the fire damages the plant power supply system and the electrical firewater pump is not available.

The calculated fuel uncovery frequency for this event is 2.3×10^{-8} per year. As in the previous cases, this estimate was based on IDCs #2, #5, #8, and #10 and on SDA #2 and SDA #3. In addition, IDC #3, the commitment to have procedures in place for communications between onsite and offsite organizations during severe weather, is also important in the analysis for increasing the likelihood that offsite organizations can respond effectively to a fire event because the availability of offsite resources increases the likelihood of recovery.

3.4.6 Heavy Load Drops

The staff investigated the frequency of a heavy load drop in or near the SFP and the potential damage to the pool from such a drop. The previous assessment done for resolution of Generic Issue 82 (in NUREG/CR-4982 (Ref. 4)) only considered the possibility of a heavy load drop on the pool wall. The assessment conducted for this study identifies other failure modes, such as the collapse of the pool floor, as also credible for some sites. Details of the heavy load assessment are given in Appendix 2C. The analysis exclusively considered drops severe enough to catastrophically damage the SFP so that pool inventory would be lost rapidly and it would be impossible to refill the pool using onsite or offsite resources. There is no possibility of mitigating the damage, only preventing it. The staff has not attempted to partition the initiator into events where there is full rapid draindown and events where there is rapid, but partial draindown. The staff assumes a catastrophic heavy load drop (creating a large leakage path in the pool) would lead directly to a zirconium fire. The time from the load drop until a fire varies depending on fuel age, burn up, and configuration. The dose rates in the pool area before any zirconium fire are tens of thousands of rem per hour, making any recovery actions (such as temporary large inventory addition) very difficult.

Based on discussions with staff structural engineers, it is assumed that only spent fuel casks are heavy enough to catastrophically damage the pool if dropped. The staff assumes a very low likelihood that other heavy loads will be moved over the SFP and that if one of these lighter loads over the SFP is dropped, it is unlikely to cause catastrophic damage to the pool.

For a non-single-failure-proof load handling system, the drop frequency of a heavy load drop is estimated, based on NUREG-0612 information, to have a mean value of 3.4×10^{-4} per year. The number of heavy load lifts was based on the NEI estimate of 100 spent fuel shipping cask lifts per year, which probably is an overestimate. For plants with a single-failure-proof load handling system or plants conforming to the NUREG-0612 guidelines, the drop frequency is estimated to have a mean value of 9.6×10^{-6} per year, again for 100 heavy load lifts per year but using data from U.S. Navy crane experience. Once the load is dropped, the analysis must then consider whether the drop significantly damages the SFP.

When estimating the failure frequency of the pool floor and pool wall, the staff assumes that heavy loads travel near or over the pool approximately 13 percent of the total path lift length (the path lift length is the distance from where the load is lifted to where it is placed on the pool floor). The staff also assumes that the critical path (the fraction of total path the load is lifted high enough above the pool to damage the structure in a drop) is approximately 16 percent. The staff estimates the catastrophic failure rate from heavy load drops to have a mean value of 2.1×10^{-5} per year for a non-single-failure-proof system relying on electrical interlocks, fuel handling system reliability, and safe load path procedures. The staff estimates the catastrophic failure rate from heavy load drops to have a mean value of 2×10^{-7} per year for a single-failure-proof system. The staff assumes that licensees that chose the non-single-failure-proof system option in NUREG-0612 performed appropriate analyses and took mitigative actions to reduce the expected frequency of catastrophic damage to the same range as for facilities with a single-failure-proof system.

NEI has made a commitment (IDC #1) for the nuclear industry that future decommissioning plants will comply with Phases I and II of the NUREG-0612 guidelines. Consistent with this industry commitment, the additional assurance of a well-performed and implemented load drop analysis, including mitigative actions, is assumed in order to arrive at an accident frequency for non-single-failure-proof systems that is comparable to the frequency for single-failure-proof systems.

SDA #5 Load Drop consequence analyses will be performed for facilities with non-single-failure-proof systems. The analyses and any mitigative actions necessary to preclude catastrophic damage to the SFP that would lead to a rapid pool draining should be performed with sufficient rigor to demonstrate that there is high confidence in the facility's ability to withstand a heavy load drop.

Although this study focuses on the risk associated with wet storage of spent fuel during decommissioning, the staff has been alert to any implications for the storage of spent fuel during power operation. With regard to power operation, the resolution of Generic Issue (GI) 82, "Beyond Design Basis Accidents in Spent Fuel Pools," and other studies of operating reactor SFPs concluded that existing requirements for operating reactor SFPs are sufficient. In developing the risk assessment for decommissioning plants, the staff evaluated the additional issue of a drop of a cask on the SFP floor rather than just on a SFP wall. As noted above, because the industry has committed to Phase II of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36," this is not a concern for decommissioning reactors.

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Operating reactors are not required to implement Phase II of NUREG-0612. The risk for SFPs at operating plants is limited by a lower expected frequency of heavy load lifts then at decommissioning plants. Nonetheless, this issue will be further examined as part of the Office of Nuclear Regulatory Research's prioritization of Generic Safety Issue 186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants," which was accepted in May 1999.

3.4.7 Spent Fuel Pool Uncovery Frequency at Times Other Than 1 Year After Shutdown

The staff has considered how changes in recovery time available to fuel handlers at 2 months, 2 years, 5 years, and 10 years after shutdown (see Table 2.1) change the insights or bottom-line numerical results from the risk assessment . The different recovery times primarily affect the human reliability analysis (HRA) results and insights. Even at 2 months after shutdown, the HRA failure estimates are small and are dominated by institutional factors (e.g., training, quality of procedures, staffing). It is therefore expected that the total fuel uncovery frequency at 2 months will continue to be dominated by the seismic contribution. At periods beyond 1 year, the increased recovery time (from a very long time to an even longer time) lowers the uncertainty that these HRA estimates really are very small, but the increased time has not translated into significant changes in the bottom-line numerical estimates because quantification of the effect of such extensions on organizational problems is beyond the state-of-the-art.

3.5 Beyond Design Basis Spent Fuel Pool Accident Scenarios (External Events)

In the following sections, the staff explains how each of the external event initiators was modeled, discusses the frequency of fuel uncovery associated with the initiator, and describes the most important insights regarding risk reduction strategies for each initiator.

3.5.1 Seismic Events

The staff performed a simplified seismic risk analysis in its June 1999 preliminary draft risk assessment to gain initial insights on seismic contribution to SFP risk. The analysis indicated that seismic events could not be dismissed on the basis of a simplified bounding approach. The additional efforts by the staff to evaluate the seismic risk to SFPs are addressed here and in Appendix 2B.

SFP structures at nuclear power plants should be seismically robust. They are constructed of thick, reinforced concrete walls and slabs lined with stainless steel liners 1/8 to 1/4 inch thick.[9] Pool walls are about 5 feet thick and the pool floor slabs are around 4 feet thick. The overall pool dimensions are typically about 50 feet long by 40 feet wide and 55 to 60 feet high. In boiling-water reactor (BWR) plants, the pool structures are located in the reactor building at an elevation several stories above the ground. In pressurized-water reactor (PWR) plants, the SFP structures are outside the containment structure and supported on the ground or partially embedded in the ground. The location and supporting arrangement of the pool structures affect

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[9]Except at Dresden Unit 1 and Indian Point Unit 1, have no liner plates. The plants were permanently shut down more than 20 years ago and no safety significant degradation of the concrete pool structure has been reported.

their capacity to withstand seismic ground motion beyond their design basis. The dimensions of the pool structure are generally derived from radiation shielding considerations rather than seismic demand needs. Spent fuel structures at nuclear power plants are able to withstand loads substantially beyond those for which they were designed.

To evaluate the risk from a seismic event at an SFP, one needs to know both the likelihood of seismic ground motion at various acceleration levels (i.e., seismic hazard) and the conditional probability that a structure, system, or component (SSC) will fail at a given acceleration level (i.e., the fragility of the SSC). These can be convolved mathematically to arrive at the likelihood that the SFP will fail from a seismic event. In evaluating the effect of seismic events on SFPs, it became apparent that although information was available on seismic hazard for nuclear power plant sites, the staff did not have fragility analyses of the pools, nor generally did licensees. The staff recognized that many of the SFPs and the buildings housing them were designed by different architect-engineers. Additionally, the pools were built to different standards as the rules changed over the years.

To compensate for the lack of knowledge of the capacity of the SFPs, the staff proposed the use of a seismic checklist during stakeholder interactions, and in a letter dated August 18, 1999, NEI proposed a checklist that could be used to show an SFP would retain its structural integrity at a peak spectral acceleration of about 1.2 g. This value was chosen, in part, because existing databases that could be used in conjunction with the checklist only go up to 1.2 g peak spectral acceleration. The checklist was reviewed and enhanced by the staff (see Appendix 2B). The checklist includes elements to assure there are no weaknesses in the design or construction or any service-induced degradation of the pools that would make them vulnerable to failure during earthquake ground motions that exceed their design-basis ground motion but are less than the 1.2 g peak spectral acceleration. The staff used a simplified, but slightly conservative method to estimate the annual probability of a zirconium fire due to seismic events and site-specific seismic hazard estimates (see Appendix 2B, Attachment 2). These calculations resulted in a range of frequencies from less than $1x10^{-8}$ per year to over $1x10^{-5}$ per year, depending on the site and the seismic estimates used.

Figures 3.2 and 3.3 show the estimated annual probabilities of a zirconium fire from a seismic event in ascending order. Figure 3.2 shows the results of convolving the site-specific LLNL seismic hazard estimates (from NUREG-1488, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains," P. Sobel, October 1993) with the generic SFP fragility analysis, and Figure 3.3 shows the results of convolving the EPRI site-specific seismic hazard estimates (Ref. 10) in a similar manner.[10] Note that the order of the sites differs somewhat in the EPRI and LLNL estimates. These figures show that for the zirconium fire frequencies using the LLNL estimates, the annual probabilities for most site clusters just above $1x10^{-6}$ per year. The mean failure probability for the sites analyzed by LLNL is about $2x10^{-6}$ per year. This value bounds 70 percent of the sites using the LLNL curves. For the EPRI curve, the mean value of the pool failure frequency is about $2x10^{-7}$ per year. In

[10] At higher accelerations, especially for plant sites east of the Rocky Mountains, there is great modeling uncertainty about the ground motions, return periods, and the possibility of cutoff. There is virtually no data at these acceleration levels.

considering these two different sets of hazard estimates, the NRC has found that both sets are reasonable and equally valid.

By passing the checklist, the SFP will be assured a high confidence with low probability of failure (HCLPF)[11] of at least 1.2 g peak spectral acceleration. The performance of the seismic checklist is identified by the staff as SDA #6.

SDA #6 Each decommissioning plant will successfully complete the seismic checklist provided in Appendix 2B to this study. If the checklist cannot be successfully completed, the decommissioning plant will perform a plant specific seismic risk assessment of the SFP and demonstrate that SFP seismically induced structural failure and rapid loss of inventory is less than the generic bounding estimates provided in this study ($<1 \times 10^{-5}$ per year including non-seismic events).

For many sites (particularly PWRs because their SFPs are closer to ground level or embedded and the motion is therefore less amplified), the plant-specific risk may be considerably lower. There are only two plant-specific SFP fragility analyses of which the staff is aware, and these were used in the analyses performed to help confirm the generic seismic capacities assumed for SFPs.

[11]The HCLPF value is defined as the peak seismic acceleration at which there is 95 percent confidence that less than 5 percent of the time the structure, system, or component will fail.

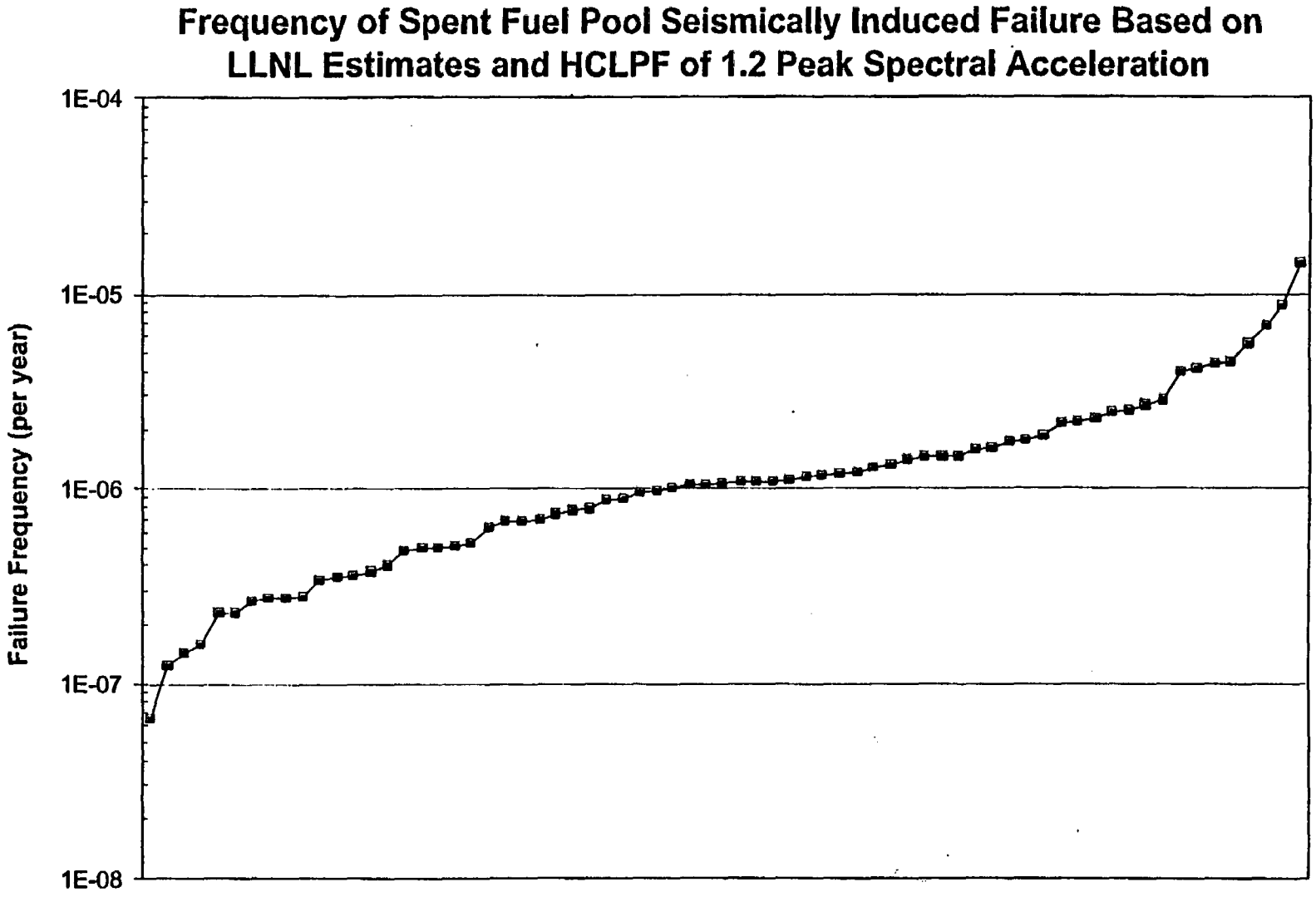

Figure 3.2

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Frequency of Spent Fuel Pool Seismically Induced Failure Based on EPRI Estimates and HCLPF of 1.2 Peak Spectral Acceleration

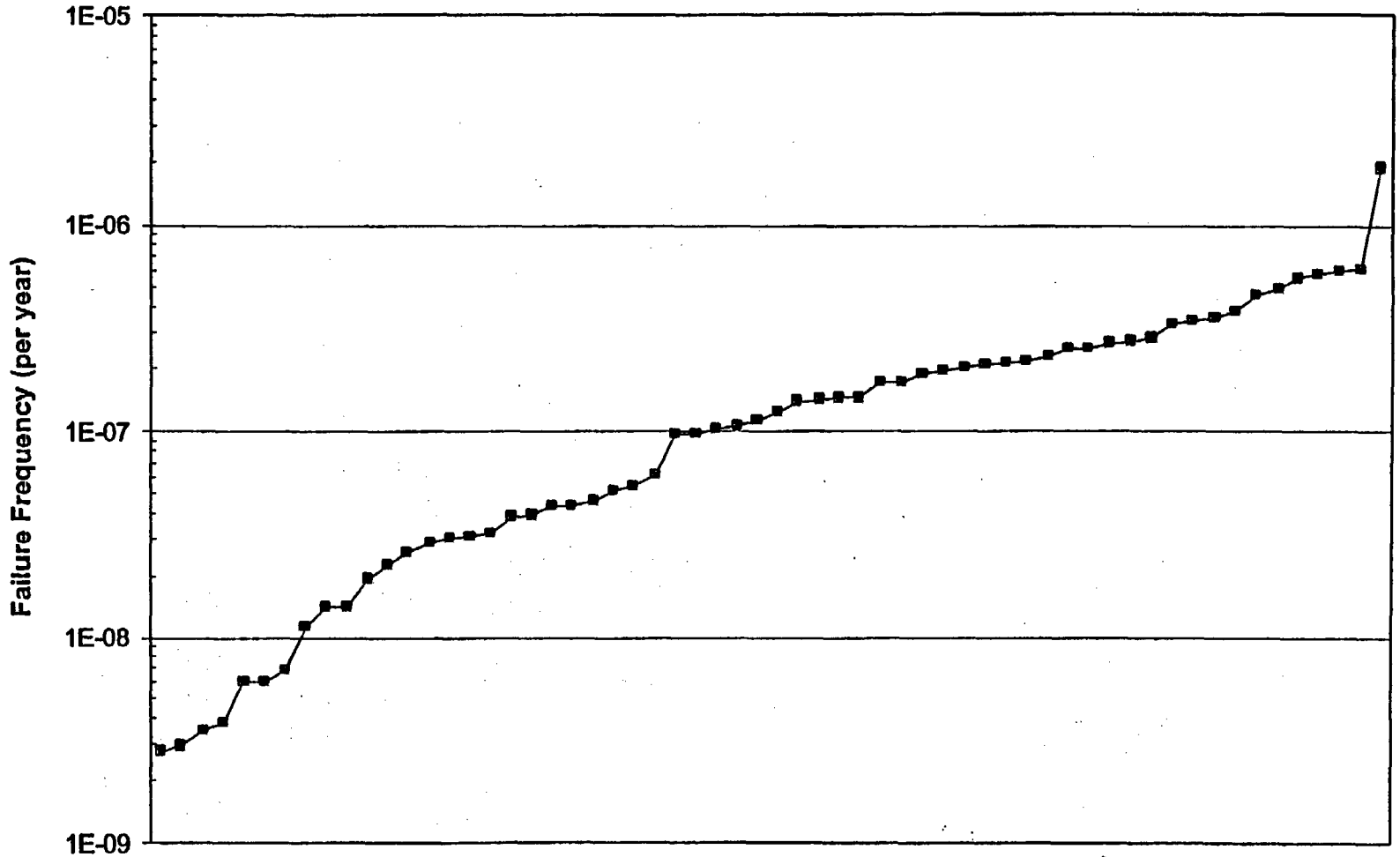

Figure 3.3

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3.5.2 Aircraft Crashes

The staff evaluated the likelihood that an aircraft crashing into a nuclear power plant site would seriously damage the spent fuel pool or its support systems (details are in Appendix 2D). The generic data provided in DOE-STD-3014-96 (Ref. 6) was used to assess the likelihood of an aircraft crash into or near a decommissioning spent fuel pool. Aircraft damage can affect the structural integrity of the spent fuel pool or the availability of nearby support systems, such as power supplies, heat exchangers, or water makeup sources, and may also affect recovery actions. There are two approaches to evaluating the likelihood of an aircraft crash into a structure. The first is the point target model, which uses the area (length times width) of the target to determine the likelihood that an aircraft will strike the target. The aircraft itself does not have real dimensions in this model. In the second approach, the DOE model modifies the point target approach to account for the wing span and the skidding of the aircraft after it hits the ground by including the additional area the aircraft could cover. The DOE model also takes into account the plane's glide path by introducing the height of the structure into the equation, which effectively increases the area of the target.

In estimating the frequency of catastrophic PWR spent fuel pool damage from an aircraft crash (i.e., the pool is so damaged that it rapidly drains and cannot be refilled from either onsite or offsite resources), the staff uses the point target area model and assumes a direct hit on a 100 x 50 foot spent fuel pool. Based on studies in NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States," it is estimated that 1 of 2 aircrafts are large enough to penetrate a 5-foot-thick reinforced concrete wall. The conditional probability that a large aircraft crash will penetrate a 5-foot-thick reinforced concrete wall is taken as 0.45 (interpolated from NUREG/CR-5042). It is further estimated that 1 of 2 crashes damage the spent fuel pool enough to uncover the stored fuel (for example, 50 percent of the time the location of the damage is above the height of the stored fuel). The estimated range of catastrophic damage to the spent fuel pool resulting in uncovery of the spent fuel is 1.3×10^{-11} to 6.0×10^{-8} per year. The mean value is estimated to be 4.1×10^{-9} per year. The frequency of catastrophic BWR spent fuel pool damage resulting from a direct hit by a large aircraft is estimated to be the same as for a PWR. Mark-I and Mark-II secondary containments generally do not appear to have any significant structures that might reduce the likelihood of aircraft penetration, although a crash into 1 of 4 sides of a BWR secondary containment may be less likely to penetrate because other structures are in the way of the aircraft. Mark-III secondary containments may reduce the likelihood of penetration somewhat, since the spent fuel pool may be protected on one side by additional structures. If instead of a direct hit, the aircraft skids into the pool or a wing clips the pool, catastrophic damage may not occur. The staff estimates that skidding aircraft are negligible contributors to the frequency of fuel uncovery resulting from catastrophic damage to the pool because skidding decreases the impact velocity. The estimated frequencies of aircraft-induced catastrophic spent fuel pool failure are bounded by other initiators.

The staff estimated the frequency of significant damage to spent fuel pool support systems (e.g., power supply, heat exchanger, makeup water supply) for three different situations. The first case is based on the DOE model including the glide path and the wing and skid area and assumes a structure 400 x 200 x 30 feet (i.e., the large building housing the support systems) with a conditional probability of 0.01 that one of these systems is hit (the critical system

occupies a 30 x 30 x 30 foot cube within the large building). This model accounts for damage from the aircraft (including, for example, being clipped by a wing). The estimated frequency range for significant damage to the support systems is 1.0×10^{-10} to 1.0×10^{-6} per year. The mean value is estimated to be 7.0×10^{-8} per year. The second case estimates the value for the loss of a support system (power supply, heat exchanger or makeup water supply). Based on the DOE model including the glide path and the wing and skid area this case assumes a 10 x 10 x 10 foot structure (i.e., the support systems are housed in a small building). The estimated frequency of support system damage ranges from 1.1×10^{-9} to 1.1×10^{-5} per year, with the mean estimated to be 7.3×10^{-7} per year. The third case uses the point model for this 10x10 structure, and the estimated value range is 2.4×10^{-12} to 1.1×10^{-8} per year, with the mean estimated to be 7.4×10^{-10} per year. Depending on the model used and the target structure size, the mean value for an aircraft damaging a support system is 7×10^{-7} per year or less. This is not the estimated frequency of fuel uncovery or a zirconium fire caused by damage to the support systems, since the frequency estimate does not include recovery, either on site or off site. As an initiator of failure of a support system leading to fuel uncovery and a zirconium fire, an aircraft crash is bounded by other more probable events. Recovery of the support system will reduce the likelihood of spent fuel uncovery.

Overall, the likelihood of significant spent fuel pool damage from aircraft crashes is bounded by other more likely catastrophic spent fuel pool failure and loss of cooling modes.

3.5.3 Tornadoes and High Winds

The staff performed a risk evaluation of tornado threats to spent fuel pools (details are in Appendix 2E). The staff assumed that very severe tornadoes (F4 to F5 tornadoes on the Fujita scale) would be required to cause catastrophic damage to a PWR or BWR spent fuel pool. These tornados have wind speeds that result in damage characterized as "devastating" or "incredible." The staff then looked at the frequency of such tornadoes and the conditional probability that if such a tornado hit the site, it would seriously damage the spent fuel pool. To do this the staff examined the frequency and intensity of tornadoes the continental United States, using the methods described in NUREG/CR-2944 (Ref. 7). The frequency of an F4 to F5 tornado is estimated to be 5.6×10^{-7} per year for the Central United States, with a U.S. average value of 2.2×10^{-7} per year.

The staff then considered what level of damage an F4 or F5 tornado could do to a spent fuel pool. Based on the buildings housing the spent fuel pools and the thickness of the spent fuel pools themselves, the conditional probability of catastrophic failure given a tornado missile is very low. Hence, the overall frequency of catastrophic pool failure caused by a tornado is extremely low (i.e., the calculated frequency of such an event is less than 1×10^{-9} per year).

It was assumed that an F2 to F5 tornado would be required to significantly damage SFP support systems (e.g., power supply, cooling pumps, heat exchanger, or makeup water supply). These tornados have wind speeds that result in damage characterized as "significant," "severe," or "worse." The frequency of an F2 to F5 tornado is estimated to be 1.5×10^{-5} per year for the Central United States, with a U.S. average value of 6.1×10^{-6} per year. This is not the estimated frequency of fuel uncovery or a zirconium fire caused by damage to the support systems, since the frequency estimate does not include recovery, either on site or off site. As an initiator of

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failure of a support system leading to fuel uncovery and a zirconium fire, a tornado is bounded by other more probable events. Recovery of the support system(s) will reduce the likelihood of spent fuel uncovery.

Missiles generated by high winds (for example, straight winds or hurricanes) are not as powerful as those generated by tornados. Therefore high winds are estimated to have a negligible impact on the frequency of catastrophic failure of the SFP resulting in fuel uncovery. Long-term loss of offsite power due to straight winds is evaluated in Section 3.4.4.

The staff estimated the frequency of significant damage to SFP support systems from straight-line winds to be very low. Damage was assumed to be caused by building collapse. Based on the construction requirements for secondary containments, the staff believes that the buildings containing BWR spent fuel pools are sufficiently robust that straight line winds will not challenge the integrity of the building. The staff assumes buildings covering PWR spent fuel pools have a concrete foundation that extends part way up the side of the building. The exterior of the rest of the building has a steel frame covered by corrugated steel siding. The PWR spent fuel buildings are assumed to be constructed to American National Standards Institute (ANSI) or American Society of Civil Engineers (ASCE) standards. Based on these assumptions, the staff believes that straight-line winds will cause buildings housing PWR spent fuel pools to fail at a frequency of 1×10^{-3} per year or less. This failure rate for support systems is subsumed in the initiating event frequency for loss of offsite power from severe weather events. The event tree for this initiator takes into account the time available for recovery of spent fuel pool cooling (approximately 195 hours for 1-year old PWR fuel and 253 hours for 1-year-old BWR fuel).

3.6 Criticality in Spent Fuel Pool

In Appendix 3, the staff performed an evaluation of the potential scenarios that could lead to criticality and identified those that are credible.

In this section the staff gives its qualitative assessment of risk due to criticality in the SFP, concluding that, with the additional assumptions, the potential risk from SFP criticality is small.

Appendix 3 references the NRC staff report "Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools." The assessment identified two credible scenarios listed below:

(1) A compression or buckling of the stored assemblies from the impact of a dropped heavy load (such as a fuel cask) could result in a more optimum geometry (closer spacing) and thus create the potential for criticality. Compression is not a problem for high-density PWR or BWR racks because they have sufficient fixed neutron absorber plates to mitigate any reactivity increase, nor is it a problem for low-density PWR racks if soluble boron is credited. But compression of a low-density BWR rack could lead to a criticality since BWR racks contain no soluble or solid neutron-absorbing material. This is not a surprising result since low-density BWR fuel racks use geometry and fuel spacing as the primary means of maintaining subcriticality. High-density racks rely on both fixed neutron absorbers and geometry to control reactivity. Low-density racks rely solely upon geometry for reactivity control. In addition, all PWR pools are borated, whereas BWR pools contain no soluble

neutron-absorbing material. If BWR pools were borated, criticality would not be possible during a low-density rack compression event.

(2) If the stored assemblies are separated by neutron absorber plates (e.g., Boral or Boraflex), loss of these plates could result in a potential for criticality for BWR pools. For PWR pools, the soluble boron in the fuel pool water is sufficient to maintain subcriticality. The absorber plates are generally enclosed by cover plates (stainless steel or aluminum alloy). The tolerances of cover plates tend to prevent any appreciable fragmentation and movement of the enclosed absorber material. The total loss of the welded cover plate is not considered feasible.

Boraflex has been found to degrade in spent fuel pools because of gamma radiation and exposure to the wet pool environment. For this reason, the NRC issued Generic Letter 96-04 on Boraflex degradation in spent fuel storage racks to all holders of operating licenses. Each addressee that uses Boraflex was requested to assess the capability of the Boraflex to maintain a 5-percent subcriticality margin and to submit to the NRC proposed actions to monitor the margin or confirm that this 5 percent margin can be maintained for the lifetime of the storage racks. Many licensees subsequently replaced the Boraflex racks in their pools or reanalyzed the criticality aspects of their pools, assuming no reactivity credit for Boraflex.

Other potential criticality events, such as events involving loose pellets or the impact of water (adding neutron moderation) during personnel actions in response to accidents, were discounted because the basic physics and neutronic properties of the racks and fuel would prevent criticality conditions from being reached with any credible likelihood. For example, without moderation fuel at current enrichment limits (no greater than 5 wt% U-235) cannot achieve criticality, no matter what the configuration. If it is assumed that the pool water is lost, a reflooding of the storage racks with unborated water may occur during personnel actions. However, both PWR and BWR storage racks are designed to remain subcritical if moderated by unborated water in the normal configuration. Thus, the only potential credible scenarios are the two scenarios described above, which involve crushing of fuel assemblies in low-density racks or degradation of Boraflex over long periods in time. These conclusions assume present light-water uranium oxide reactor fuel designs. Alternative fuel designs, such as mixed oxide (MOX) fuels will have to be reassessed to ensure that additional vulnerabilities for pool criticality do not exist.

To gain qualitative insights on credible criticality events, the staff considered the sequences of events that must occur. For scenario 1, a heavy load drop into a low-density racked BWR pool, compressing the assemblies would be required. From its analysis of the heavy load drop documented in Appendix 2C, the staff has determined the likelihood of a heavy load drop from a single failure-proof crane to have a mean frequency of approximately $9.6X10^{-6}$ per year, assuming 100 cask movements per year at the decommissioning facility. From the load path analysis done in Appendix 2C, the staff estimates that the load is over or near the pool approximately 13 percent of the movement path length, depending on the plant's layout. The additional frequency reduction in the appendix to account for the fraction of time that the heavy load is lifted high enough to damage the pool liner is not applicable here because the fuel assemblies can be crushed by a smaller impact velocity than required to need to crush the pool

liner. Therefore, the staff estimates that the potential initiating frequency for crushing is approximately $1.2X10^{-6}$ per year (based upon 100 lifts per year). The criticality calculations in Appendix 3 show that even if the low-density BWR assemblies were crushed by a transfer cask, it is "highly unlikely" that a configuration would be produced that would result in a severe reactivity event, such as a steam explosion that could damage and drain the spent fuel pool. The staff judges the chances of such a criticality event to be well below 1 chance in 100 even if the transfer cask drops directly onto the assemblies. This would put the significant criticality likelihood well below $1X10^{-8}$ per year.

Deformation of the low-density BWR racks by the dropped transfer cask was shown to most likely not result in any criticality events. However, if some mode of criticality was to be induced by the dropped transfer cask, it would likely be a small return to power for a very localized region, rather than the severe response discussed in the paragraph above. This type of event would have essentially no offsite (or onsite) consequences since the heat of the reaction would be removed by localized boiling in the pool, and water would shield the site operating staff. The reaction could be terminated with relative ease by the addition of boron to the pool. Therefore, the staff believes that qualitative (as well as some quantitative) assessment of scenario 1 demonstrates that it poses no significant risk to the public from SFP operation while the fuel remains stored in the pool.

With respect to scenario 2 (the gradual degradation of the Boraflex absorber material in high-density storage racks), there is currently insufficient data to quantify the likelihood of criticality due to the degradation. However, the current programs in place at operating plants to assess the condition of the Boraflex and take remedial action if necessary provide sufficient confidence that pool reactivity requirements will be satisfied. In order to meet the RG 1.174 safety principle of maintaining sufficient safety margins, the staff judges that continuation of such programs into the decommissioning phase should be considered at all plants until all high-density racks are removed from the SFP. As such, SDA #7 should be considered in future regulatory activities associated with SFP requirements. This additional assumption is identified as SDA #7.

SDA #7 Licensees will maintain a program to provide surveillance and monitoring of Boraflex in high-density spent fuel racks until such time as spent fuel is no longer stored in these high-density racks.

Based upon the above conclusions and the staff decommissioning assumption, the staff believes that qualitative risk insights demonstrate conclusively that SFP criticality poses no meaningful risk to the public.

3.7 Consequences and Risks of SFP Accidents

This section assesses the consequences and risks associated with SFP accidents. The consequences are assessed in Section 3.7.1. Results are provided for both early evacuation and late evacuation cases[12] to address the impact of evacuation on consequences, and for two

[12] Early evacuation is initiated and completed before the SFP release. Late evacuation is not completed before release.

different source terms to show the impact of source term uncertainties on results. In Section 3.7.2, the severe accident consequences for either the early or late evacuation cases are assigned to each of the major types of SFP accidents, as appropriate, and then combined with the respective event frequencies to provide a scoping estimate of SFP risks. The risks of SFP accidents are shown to meet the Commission's safety goals. The impact of changes in EP regulations on these risk measures is discussed later in Section 4.

3.7.1 Consequences of SFP Accidents

Earlier analyses in NUREG/CR-4982, "Severe Accidents in Spent Fuel Pools in Support of Generic Issue 82," and NUREG/CR-6451, "A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants," included a limited analysis of the offsite consequences of a severe SFP accident occurring up to 90 days after the last discharge of spent fuel into the SFP. The analysis showed that the consequences of an SFP accident could be comparable to those for a severe reactor accident. As part of its effort to develop generic, risk-informed requirements for decommissioning, the staff performed a further analysis of the offsite radiological consequences of beyond-design-basis SFP accidents. Varying the evaluation and other modeling assumptions, the staff performed an initial set of calculations to extend the earlier analyses to SFP accidents occurring 1 year after plant shutdown and to supplement the earlier analyses with additional sensitivity studies. The results of these calculations were documented in the February 2000 study, and are provided in Appendix 4.

Subsequently, the ACRS raised issues with the source term and plume modeling for SFP accidents. In particular, the ACRS believed that the ruthenium and fuel fines releases were too low and the plume was too narrow. To address these issues, the staff performed additional sensitivity studies, as documented in Appendix 4A of this study.

To provide insight into the impact on results of decay times shorter or longer than 1 year, additional consequence calculations were performed using fission product inventories at 30 and 90 days and 2, 5, and 10 years after final shutdown. The results of these consequence calculations were used as the basis for assessing the risk from SFP accidents. These results are summarized in Tables 3.7-1 and 3.7-2 for several key consequence measures, and are described in more detail in Appendix 4B. These consequences are conditional upon the occurrence of an accident that results in an SFP fire, i.e., the consequences are on a "per event" rather than a "per year," basis and do not account for the probability of the event.

These calculations were based on the Surry site, although the SFP accident consequences could be greater at higher population sites, the quantitative health objectives used in comparisons to the Commission's Safety Goals (see Section 3.7.3) represent risk to the average individual within 1 mile and 10 miles of the plant, and should be relatively insensitive to the site specific population.

Table 3.7-1 Consequences of an SFP Accident With a High Ruthenium Source Term (per event)

Time After Shutdown	Mean Consequences for High Ruthenium Source Term (Surry population, 95% evacuation)			
	Early Fatalities	Societal Dose (p-rem within 50 miles)	Individual Risk* of Early Fatality (within 1mile)	Individual Risk* of Latent Cancer Fatality (within 10 miles)
Late Evacuation				
30 days	192	2.37×10^{7}	4.43×10^{-2}	8.24×10^{-2}
90 days	162	2.25×10^{7}	4.19×10^{-2}	8.20×10^{-2}
1 year	77	1.93×10^{7}	3.46×10^{-2}	8.49×10^{-2}
2 years	19	1.69×10^{7}	2.57×10^{-2}	8.42×10^{-2}
5 years	1	1.45×10^{7}	8.96×10^{-2}	7.08×10^{-2}
10 years	-	1.34×10^{7}	4.68×10^{-2}	6.39×10^{-2}
Early Evacuation				
30 days	7	1.35×10^{7}	2.01×10^{-3}	4.79×10^{-3}
90 days	4	1.29×10^{7}	1.87×10^{-3}	4.77×10^{-3}
1 year	1	1.12×10^{7}	1.50×10^{-3}	4.33×10^{-3}
2 years	-	9.93×10^{6}	1.12×10^{-3}	3.70×10^{-3}
5 years	-	8.69×10^{6}	3.99×10^{-4}	2.93×10^{-3}
10 years	-	8.13×10^{6}	2.05×10^{-4}	2.64×10^{-3}

* Conditional on event - Total frequency for all events is shown in Table 3.1 as less than 3×10^{-6} per year.

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Table 3.7-2 Consequences of an SFP Accident With a Low Ruthenium Source Term (per event)

Time After Shutdown	Mean Consequences for Low Ruthenium Source Term (Surry population, 95% evacuation)			
	Early Fatalities	Societal Dose (p-rem within 50 miles)	Individual Risk* of Early Fatality (within 1mile)	Individual Risk* of Latent Cancer Fatality (within 10 miles)
Late Evacuation				
30 days	2	5.58×10^{6}	1.27×10^{-2}	1.88×10^{-2}
90 days	1	5.43×10^{6}	9.86×10^{-3}	1.82×10^{-2}
1 year	1	5.28×10^{6}	7.13×10^{-3}	1.68×10^{-2}
2 years	-	5.12×10^{6}	5.64×10^{-3}	1.58×10^{-2}
5 years	-	4.90×10^{6}	3.18×10^{-3}	1.43×10^{-2}
10 years	-	4.72×10^{6}	1.63×10^{-3}	1.29×10^{-2}
Early Evacuation				
30 days	-	4.12×10^{6}	8.36×10^{-4}	9.92×10^{-4}
90 days	-	4.02×10^{6}	6.83×10^{-4}	9.62×10^{-4}
1 year	-	3.95×10^{6}	5.44×10^{-4}	9.09×10^{-4}
2 years	-	3.87×10^{6}	4.41×10^{-4}	8.71×10^{-4}
5 years	-	3.77×10^{6}	2.54×10^{-4}	8.14×10^{-4}
10 years	-	3.69×10^{6}	1.47×10^{-4}	7.70×10^{-4}

* Conditional on event - Total frequency for all events is shown in Table 3.1 as less than 3×10^{-6} per year.

The consequences in Table 3.7-1 are based on the upper bound source term described in Appendix 4B. With the exception of ruthenium and fuel fines, the release fractions are from NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Ref. 1), and include the ex-vessel and late in-vessel phase releases. The ruthenium release fraction is for a volatile fission product in an oxidic (rather than metallic) form. This is consistent with the experimental data reported in Reference 8. The source term is considered to be bounding for several reasons. First, rubbling of the spent fuel after heatup to about 2500 °K is expected to limit the potential for ruthenium release to a value less than that for volatile fission products. Second, following the Chernobyl accident, ruthenium in the environment was found to be in the metallic form (Ref. 2). Metallic ruthenium (Ru-106) has about a factor of 50 lower dose conversion factor (rem per Curie inhaled) than the oxidic ruthenium assumed in the Melcor Accident Consequence Code System (MACCS) calculations. Finally, the fuel fines release fraction is that from the Chernobyl accident (Ref. 3). This is considered to be bounding because the Chernobyl accident involved more extreme conditions (i.e., two explosions followed

by a prolonged graphite fire) than an SFP accident. In subsequent discussions, this source term is referred to as the high ruthenium source term.

The consequences obtained using the source term in NUREG-1465 (which treats ruthenium as a less volatile fission product) in conjunction with SFP fission product inventories are provided in Table 3.7-2 for comparison. In subsequent discussions, this source term is referred to as the low ruthenium source term.

The consequence calculations for both the high and low ruthenium source terms assume that all of the fuel assemblies discharged in the final core off-load and the previous 10 refueling outages participate in the SFP fire. These assemblies are equivalent to about 3.5 reactor cores. Approximately 85 percent of all the ruthenium in the pool is in the last core off-loaded since the ruthenium-106 half-life is about 1 year. For cesium-137, with a 30-year half-life, the inventory decays very slowly and is abundant in all of the batches considered. The staff assumed that the number of fuel assemblies participating in the SFP fire remains constant and did not consider the possibility that fewer assemblies might be involved in an SFP fire in later years because of substantially lower decay heat in the older assemblies. Based on the limited analyses performed to date, fire propagation is expected to be limited to less than two full cores 1 year after shutdown (see Appendix 1A). Thus, the assumption that 3.5 cores participate adds some conservatism to the calculation of long-terms effects associated with cesium, but is not important with regard to the effects of ruthenium.

The results for early fatality and societal dose (person-rem) consequences for an SFP accident are graphed in Figures 3.7-1 and 3.7-2. The early fatality plots are truncated at a value of one early fatality since fractions of a fatality are not meaningful. Since no early fatalities were predicted for the low ruthenium source term with early evacuation, a curve is not shown for that case in Figure 3.7-1. Because latent cancer fatalities are directly proportional to societal dose through a dose-to-cancer-risk conversion factor within the MACCS2 consequence code (Ref. 9), results for latent cancer fatalities are not displayed separately.

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Early Fatality Consequences for Spent Fuel Pool Source Terms

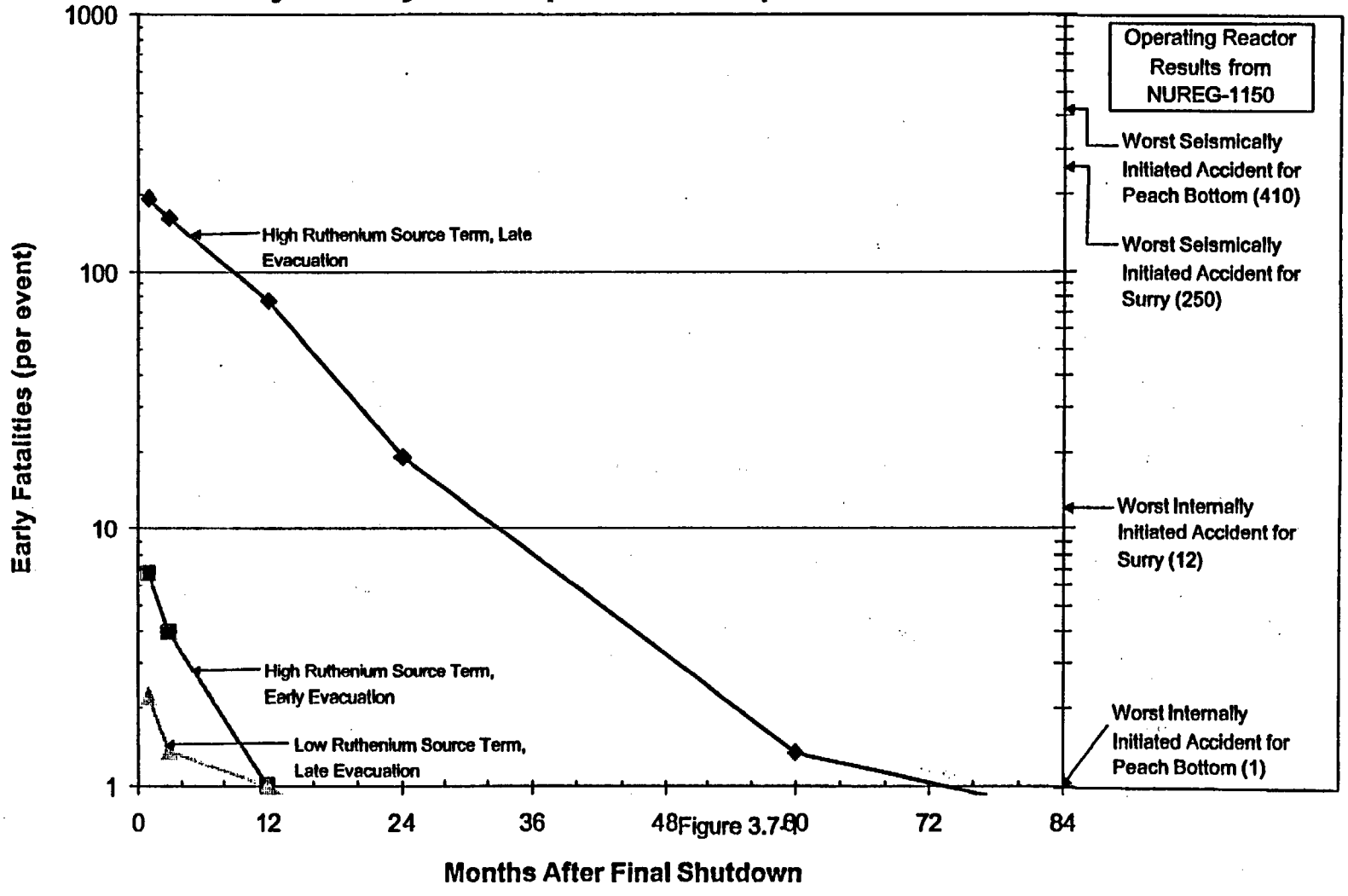

Figure 3.7

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Societal (Person-rem) Consequences for Spent Fuel Pool Source Terms

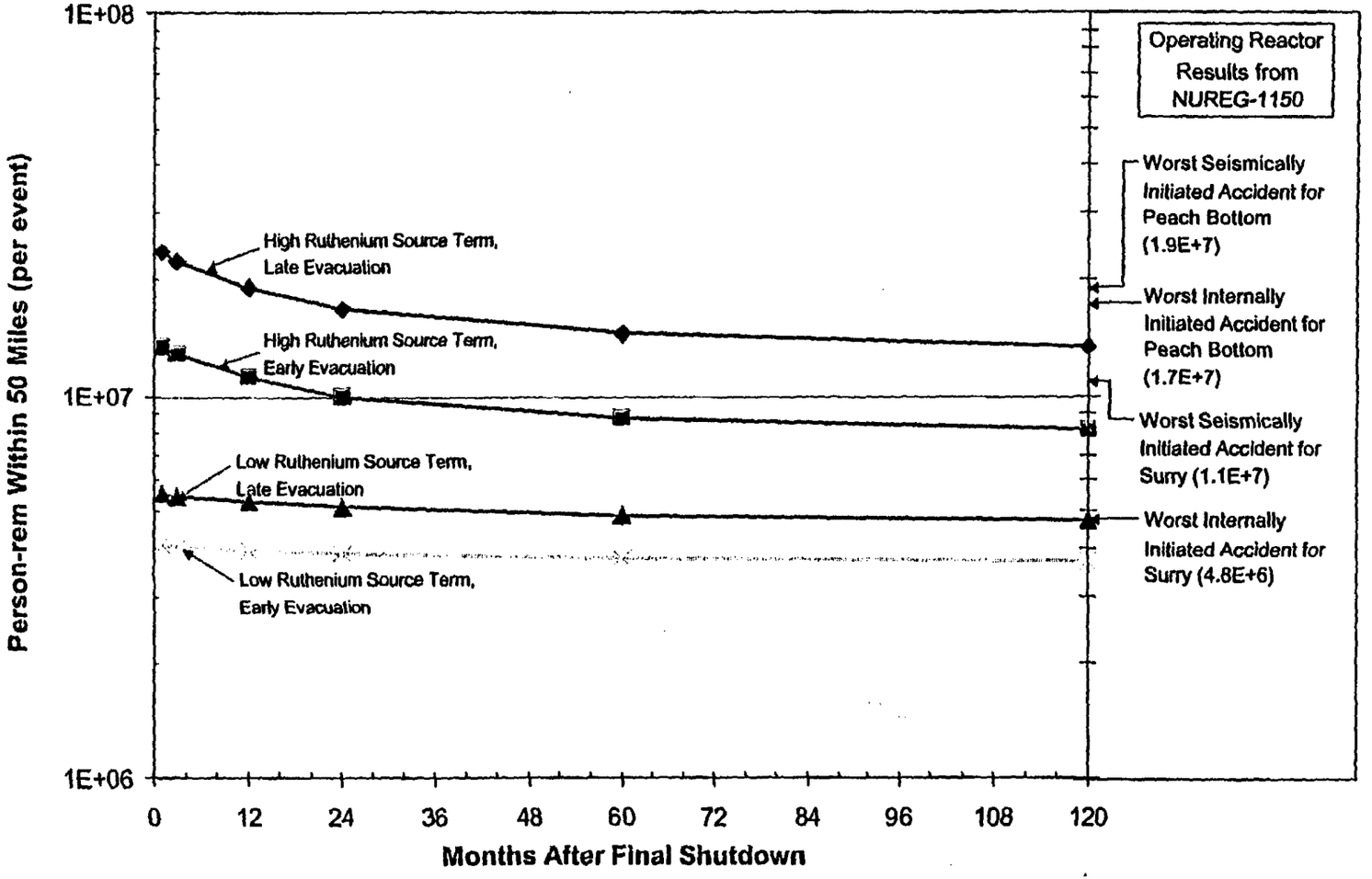

Figure 3.7-2

Consequence estimates are also included on Figures 3.7-1 and 3.7-2 for the two operating reactors for which risk results for both internal and seismic events are available in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," and the supporting NUREG/CR-4551 reports, "Evaluation of Severe Accident Risks: Surry Unit 1" and "Evaluation of Severe Accident Risks: Peach Bottom, Unit 2." The values shown are for the reactor accident source terms that produced the greatest number of early fatalities (Figure 3.7-1) or the greatest societal dose and latent cancer fatalities (Figure 3.7-2). Results are displayed separately for internally and seismically initiated accidents and indicate that for these plants, reactor accident consequences for seismically initiated events are substantially higher than those for internally initiated events. Although the consequences for the high ruthenium source term diminish more quickly than for the low ruthenium source term, these curves do not converge because of the long half-lives of the fuel fines in the high ruthenium source term.

An examination of Figure 3.7-1 indicates the following:

- Early fatality consequences for spent fuel pool accidents can be as large as for a severe reactor accident even if the fuel has decayed several years. This is attributable to the significant health effect of ruthenium, and the ruthenium-106 half-life of about 1 year. There is also an important but lesser contribution from cesium.

- A large ruthenium release fraction is important to consequences, but not more important than the consequences of a reactor accident large early release.

- The effect of early evacuation (if possible) is to offset the effect of a large ruthenium release fraction. This effect is comparable to that for reactor accidents.

- For the low ruthenium source term, no early fatality is expected after 1 year decay even with late evacuation.

For the longer term consequences Figure 3.7-2 indicates:

- Long-term consequences remain significant as long as a fire is possible. These consequences are due primarily to the effect of cesium-137, which remains abundant even in significantly older fuel because of its long (30-year) half-life. Ruthenium and evacuation have notable long-term consequences but do not change the conclusion.

3.7.2 Risk Modeling for SFP Accidents

The quantitative assessment of risk involves combining the estimated frequencies of severe accident sequences with their corresponding offsite consequences. In this section, severe accident consequences reported in Tables 3.7-1 and 3.7-2 are assigned to each of the major types of events that lead to uncovery of the spent fuel, and then combined with the respective event frequencies to provide a scoping estimate of SFP risks.

The SFP accidents discussed in Section 3 can be broadly classified as either boildown or rapid draindown sequences. Rapid draindown sequences are further divided into seismically- and

non-seismically-initiated events. In assigning consequences to each of these events, the staff considered whether protective measures to evacuate the population around the site could be effectively implemented before fission product release. This included consideration of the effectiveness of offsite notification, the delay between event initiation and fission product release (dependent on time after shutdown), the time required to initiate and complete an evacuation, and the impact that a relaxation in current emergency planning requirements might have on these factors. As a result of this assessment, consequences were assigned based on either the early evacuation case or late evacuation case.

The frequency and consequence modeling is briefly described below for each type of SFP accident. The resulting risk estimates for each sequence (in terms of early fatalities and societal dose per year) are presented in Figures 3.7-3 through 3.7-6 and discussed in Section 3.7.3.

Boil Down Sequences

Boil down sequences (including loss of inventory events) and their associated frequencies are listed in Table 3.7-3. These sequences involve heatup of the pool to boiling followed by gradual reduction in pool level until the spent fuel is eventually uncovered. This process would take over 100 hours at 60 days, and substantially longer at later times as shown in Table 2.1. The long delay provides sufficient time for licensee staff to effectively intervene in the large majority of these events, and results in very low frequencies of fuel uncovery. For those events that proceed to fuel uncovery, fuel heatup will continue until either steady-state conditions are achieved or cladding oxidation occurs. All boil down sequences that uncover spent fuel were assumed to result in an SFP fire. Loss of inventory events are classified as boil down events since the time to uncover the fuel will be in excess of 24 hours (as described in Section 4.5.4.1 of Appendix 2A) and will provide ample time for licensee to take corrective measures.

Table 3.7-3Frequency of Boil Down Events Leading to Spent Fuel Uncovery (for times greater than 60 days after shutdown)

Initiating Event	Frequency (per year)
Loss of offsite power—severe weather	1.1×10^{-7}
Loss of offsite power—plant-centered and grid-related events	2.9×10^{-8}
Internal fire	2.3×10^{-8}
Loss of pool cooling	1.4×10^{-8}
Loss of coolant inventory	3.0×10^{-9}
Total	1.8×10^{-7}

The failure paths leading to a zirconium fire involve failure to acquire offsite resources to makeup pool inventory, despite the large amount of time available for recovery in the boildown event. For sequences involving loss of offsite power due to severe weather, the weather is assumed to drain regional resources or limit access to the facility. The staff reasoned that if it

is difficult for offsite resources to reach the facility or if regional resources are engaged in other efforts, then it would also be unlikely that the population in the area would be effectively notified and/or evacuated under these conditions. For sequences other than loss of off-site power due to severe weather, the dominant reason that recovery is not provided in the failure paths is a general breakdown in the overall facility organization. The failure to acquire offsite resources also implies a failure to contact regional authorities and declare an emergency when the SFP level drops below the proceduralized limit in these sequences. Accordingly, the consequences for boildown sequences are based on results for the late evacuation case (Tables 3.7-1 and 3.7-2). This same reasoning is applied for cases with and without EP relaxations and for all times after shutdown. The net effect is that EP, as well as relaxations in EP, do not impact the risk associated with those boildown sequences that proceed to spent fuel uncovery.

Rapid Draindown Due to Seismic Events

Given the robust structural design of SFPs, it is expected that a seismic event with peak spectral acceleration several times larger than the safe shutdown earthquake (SSE) would be required to produce catastrophic failure of the structure. The estimated frequency of events of this magnitude differs greatly among experts and is driven by modeling uncertainties. The estimated frequency of seismic events sufficiently large to result in structural failure of the SFP is given in Table 3.7-4 and is based on the LLNL and EPRI seismic hazard estimates.

Both the LLNL and EPRI hazard estimates were developed as best estimates and are considered valid by the NRC. Furthermore, because both sets of curves are based upon data extrapolation and expert opinion, there is no technical basis for excluding consideration of either set.

Using the LLNL hazard estimates, a return frequency equivalent to the pool performance guideline ($1x10^{-5}$ per year) for a 1.2g peak spectral acceleration (PSA) ground motion bounds all but four sites (one Central and Eastern and three Western U.S. sites). The frequency for the remaining sites falls in the range of less than $7x10^{-8}$ per year to $9x10^{-6}$ per year. The majority (45 sites) have hazard estimates (for a 1.2 PSA ground motion) near $1x10^{-6}$ per year and 20 sites fall below $6x10^{-7}$ per year. The mean value for the population of plants is approximately $2x10^{-6}$ per year.

If EPRI hazard estimates were used, only one site would have an estimate that exceeds $1x10^{-6}$ per year (excluding Western sites).[13] Ten sites are near $5x10^{-7}$ per year, and the remaining 49 sites analyzed by EPRI have estimates less than $3x10^{-7}$ per year, with half of these sites (25 sites) estimated at less than $7x10^{-8}$ per year. The mean value for the population of plants is approximately $2x10^{-7}$ per year.

In characterizing the risk of seismically induced SFP accidents for the population of sites, the staff has displayed results based on both the LLNL and the EPRI hazard estimates, and has used an accident frequency corresponding to the mean value for the respective distributions,

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[13]EPRI seismic estimates were not developed for all sites east of the Rocky Mountains. Six sites have LLNL but no EPRI hazard estimates.

i.e., a frequency of $2x10^{-6}$ per year to reflect the use of the LLNL hazard estimates and a frequency of $2x10^{-7}$ per year to reflect use of the EPRI hazard estimates. Use of the mean value facilitates comparisons with the Commission's quantitative safety goals and quantitative health objectives (QHOs). About 70 percent of the sites are bounded using the mean value.

Table 3.7-4Mean Frequency of Rapid Draindown Due to Seismic Events

Source of Hazard Estimate	Frequency (per year)
LLNL	$2x10^{-6}$
EPRI	$2x10^{-7}$

Likely SFP failure modes and locations are discussed in Attachment 2 to Appendix 2B. The conclusion is that drainage of the pool would be fairly rapid and a small amount of water is likely to remain in the pool, with post-seismic-failure, water depths ranging from about zero to about 4 feet depending upon the critical failure mode. For purposes of consequence assessment, all seismically initiated sequences were assumed to result in a rapid draindown followed by an SFP fire, regardless of the SFP failure mode and location, which are plant-specific.

The SFP risk estimates are strongly dependent on the assumptions about the effectiveness of emergency evacuation in seismic events, since these events dominate the SFP fire frequency. In NUREG-1150, evacuation in seismic events was treated in either of two ways, depending on the peak ground acceleration (PGA) of the earthquake:

- For low PGA earthquakes, the population was assumed to evacuate; however, the evacuation was assumed to start later and proceed more slowly than evacuation for internally initiated events.
- For high PGA earthquakes, it was reasoned that there would be no effective evacuation and that many structures would be uninhabitable.

Since the seismic contribution to SFP fire frequency is driven by events with ground motion several times larger than the SSE, the reasoning that there would be no effective evacuation was adopted in developing the seismic contribution to the risk. This is consistent with the expert opinion provided in Attachment 2 to Appendix 2B about the expected level of collateral damage within the emergency planning zone in a seismic event large enough to cause the SFP failure. Specifically, for ground motion levels that correspond to SFP failure in the Central and Eastern United States, it is expected that electrical power would be lost and more than half of the bridges and buildings (including those housing communication systems and emergency response equipment) would be unsafe even for temporary use within at least 10 miles of the plant. This approach is also consistent with previous Commission rulings on San Onofre and Diablo Canyon in which the Commission found that for those risk-dominant earthquakes that cause very severe damage to both the plant and the offsite area, emergency response would have marginal benefit because of offsite damage.

The consequences for seismic sequences are therefore based on results for the late evacuation cases in Tables 3.7-1 and 3.7-2. The same reasoning is applied for cases with and without EP relaxations and for all times after shutdown. The net effect is that EP, as well as relaxations in EP, do not impact the risk associated with seismic events that result in SFP failure. A sensitivity study was also done to explore the impact on risk if the seismic event only partially degrades the emergency response (see Section 4.2.1).

Rapid Draindown Due to Non-Seismic Events

Non-seismically-initiated events leading to rapid draindown are listed in Table 3.7-5. These events are dominated by cask drop accidents, with the next highest contributor nearly two orders of magnitude lower.

Table 3.7-5Frequency of Rapid Draindown Spent Fuel Uncovery Due to Nonseismic Events

Initiating Event	Frequency (per year)
Cask drop	2.0×10^{-7}
Aircraft impact	2.9×10^{-9}
Tornado missile	$<1.0 \times 10^{-9}$
Total	2.0×10^{-7}

Cask drop accidents that lead to catastrophic failure of the SFP include accidents in which the load is dropped either on the pool floor or on or near the pool wall. Load drops on the pool floor are more likely to result in complete draindown of the pool and create an air flow path through the fuel assemblies. Load drops on the pool wall would likely result in residual water in the pool, which would obstruct air flow. For purposes of consequence assessment, all cask drop accidents leading to fuel uncovery were assumed to result in a rapid draindown followed by an SFP fire.

Depending upon the pool failure mode and location, the fuel could be air cooled, or heatup could be close to adiabatic as a result of air flow blockage. As discussed in Appendix 1A for either adiabatic or air flow conditions (at 60 GWD/MTU burnup), the time of fission product release would be about 4 hours for a PWR and 8 hours for a BWR for accidents initiated 1 year following shutdown. For cases with air cooling, close to 1 day is available after 3 years decay. Even with adiabatic heatup, 1 day is available after 5 years of decay. At 60 days after shutdown, fission product release could begin as early as 2 hours after fuel uncovery. The actual time would depend on reactor type, fuel burnup, fuel rack structure, and other plant-specific parameters, as discussed in Appendix 1A. The fuel handlers would be immediately aware of a cask drop accident. It is expected that with procedures that specify the SFP water level at which an emergency is to be declared, the proper offsite authorities would be promptly informed.

For the case in which current EP requirements are retained, it was assumed that cask drop accidents occurring 1 or more years following shutdown would afford sufficient time to implement protective measures before fission products were released. This is consistent with

the evacuation time estimates in the NUREG-1150 study for Surry, which assumed a 1.5 hour delay time and a 4 mile per hour evacuation speed. Thus the consequences at less than 1 year following shutdown are based on late evacuation, and the consequences at 1 year and beyond are based on early evacuation when full EP requirements are retained.

Relaxations in EP requirements are expected to result in additional delays in initiation and implementation of protective measures relative to the case in which current EP requirements are retained. If offsite preplanning requirements were to be relaxed, as many as 10 to 15 hours may be required at some sites to initiate an evacuation. Based on either air-cooled or adiabatic heatup rates for the reference pool, the minimum time to fission product release following a load drop that catastrophically damages the pool is about 8–9 hours for PWR pools and about 15 hours for BWR pools 2 years following shutdown (see Appendix 1A). These release times increase significantly by 5 years following shutdown (i.e., greater than 24 hours even with adiabatic heatup rates). For the case in which current EP requirements are relaxed, the consequences within the first 2 years following shutdown are based on late evacuation, and the consequences at 5 years and after are based on the early evacuation results reported in Tables 3.7-1 and 3.7-2.

3.7.3 Risk Results

The frequency and consequences for each SFP accident were combined to provide a scoping estimate of the risk of SFP accidents. The frequency of each event was based on the estimated value at 1 year following shutdown as described above, and was assumed to remain constant over time. In reality, the frequency would vary with time, and could be higher or lower than the 1-year estimate, as a result of plant configuration changes described in Section 3.1 (e.g., replacement of operating plant pool cooling and makeup systems with skid-mounted systems) and reductions in decay heat levels (which would impact human reliability estimates). However, as described in Section 3.4.7, these impacts are not expected to change the insights from the risk assessment for decay times greater than 60 days.

Figures 3.7-3 and 3.7-4 show the total early fatality risk and societal risk as a function of time after final shutdown. Companion curves are provided based on both the LLNL and the EPRI seismic hazard studies since both studies are considered equally valid. The SFP risk results are shown in these figures for both the high ruthenium source term and a fuel burnup of 60 GWD/MTU. Also shown are the corresponding mean risk measures for two operating plants, Surry and Peach Bottom,[14] for which risk results for both internal and seismic events are given in NUREG-1150.

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[14] The LLNL seismic risk results reported in NUREG-1150 are based on a 1989 version of the LLNL hazard estimates. An update of these estimates performed in 1993 resulted in a factor of 10 reduction in the LLNL mean hazard for Peach Bottom and a smaller reduction for Surry. To provide a more meaningful comparison, the LLNL seismic risk results for Peach Bottom reported in NUREG-1150 have been reduced by a factor of 10. The results for Surry and the EPRI seismic risk results are not affected by this adjustment.

Figures 3.7-5 and 3.7-6 show the risk contribution from cask drop events, which are the only events modeled that are significantly impacted by EP. For the case in which current EP requirements are retained, the consequences at 1 year and beyond are based on early evacuation (the lower, solid curve). For the case in which current EP requirements are relaxed, the consequences within the first 2 years following shutdown are based on late evacuation (the upper, solid curve), and the consequences at 5 years and beyond are based on early evacuation, as discussed in Section 3.7.2.

000162

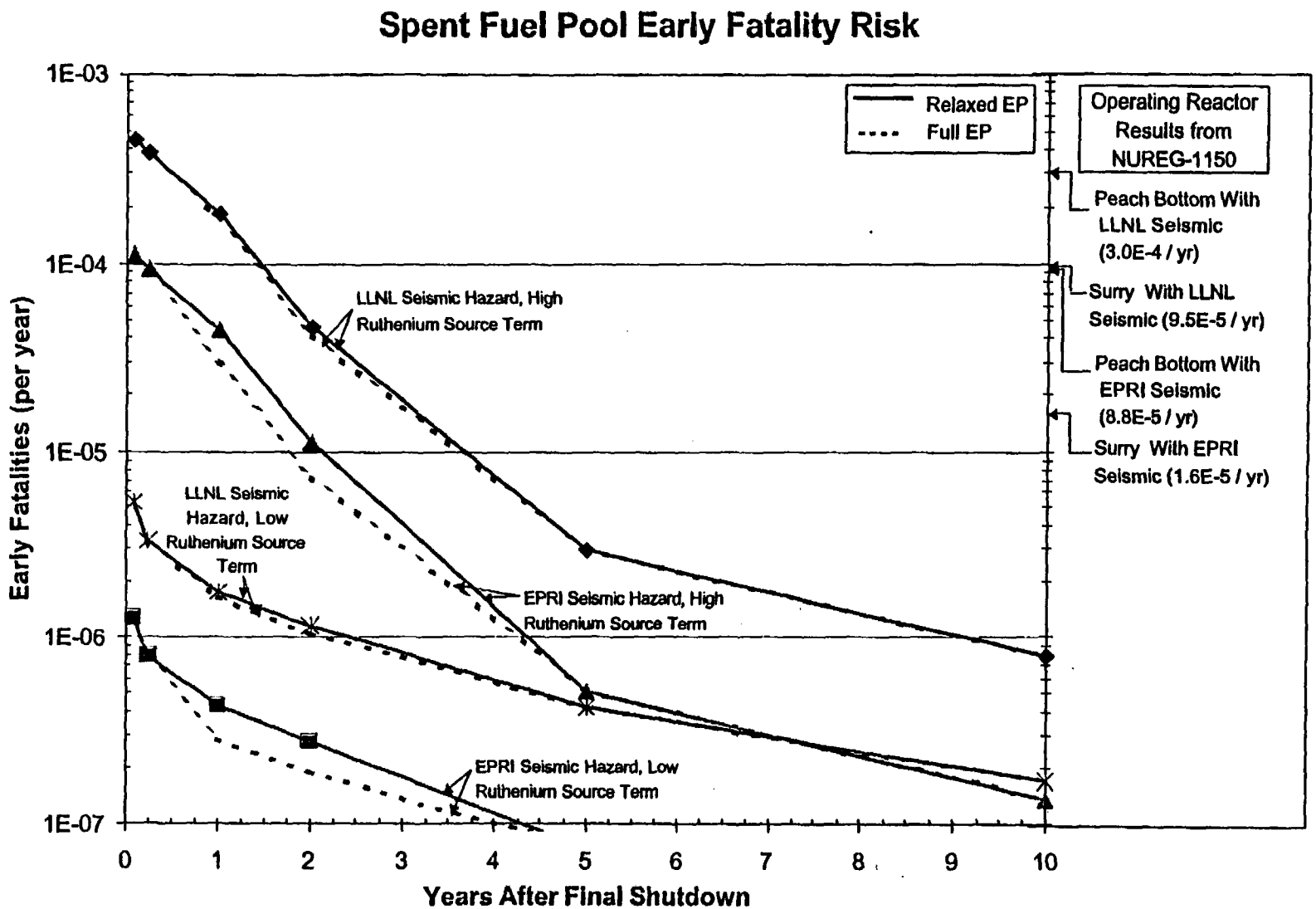

Figure 3.7-3

000170

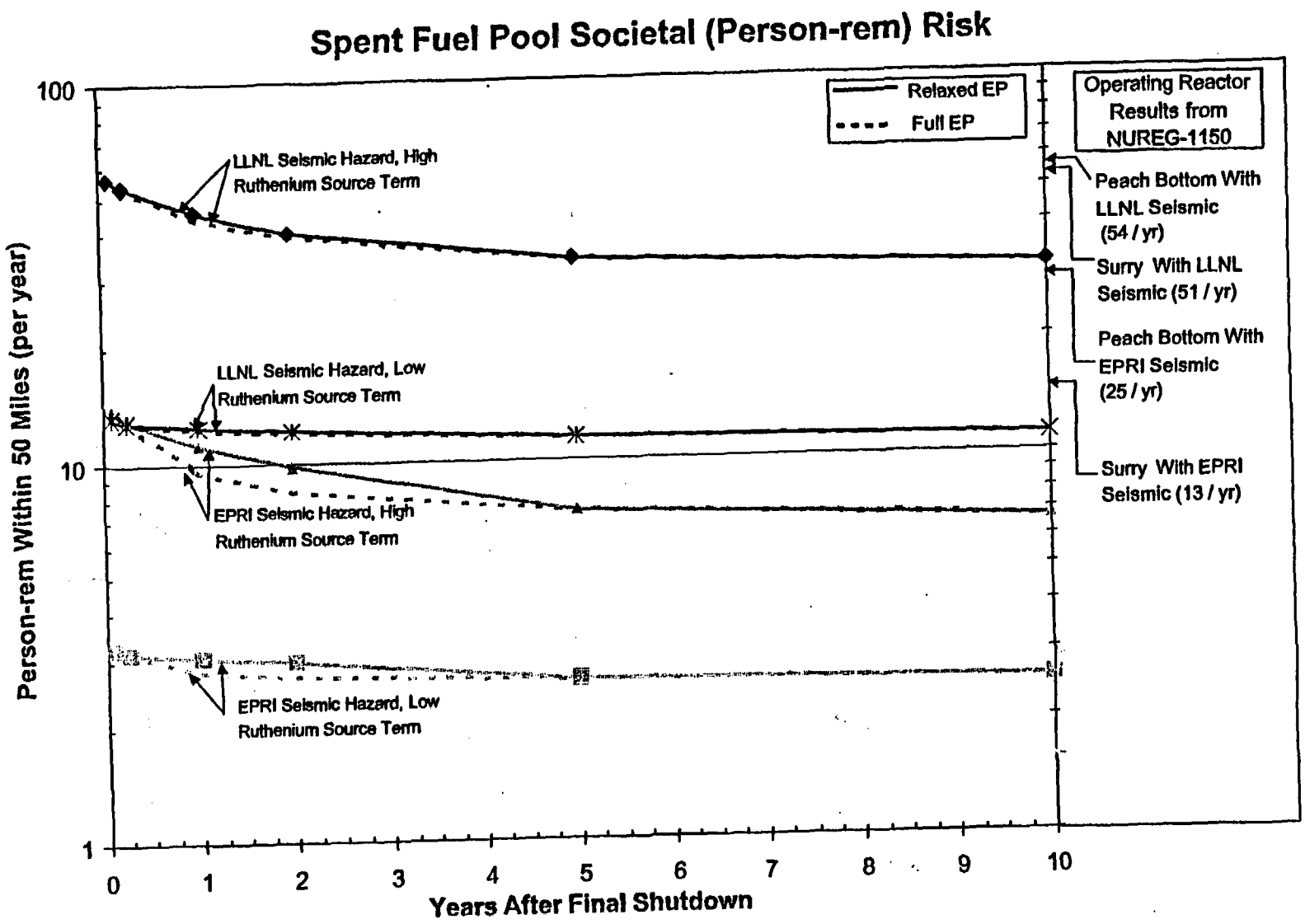

Figure 3.7-4

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Sensitivity of Early Fatality Risk to Emergency Planning -- Cask Drop Event

(Conditional upon: High Ruthenium Source Term)

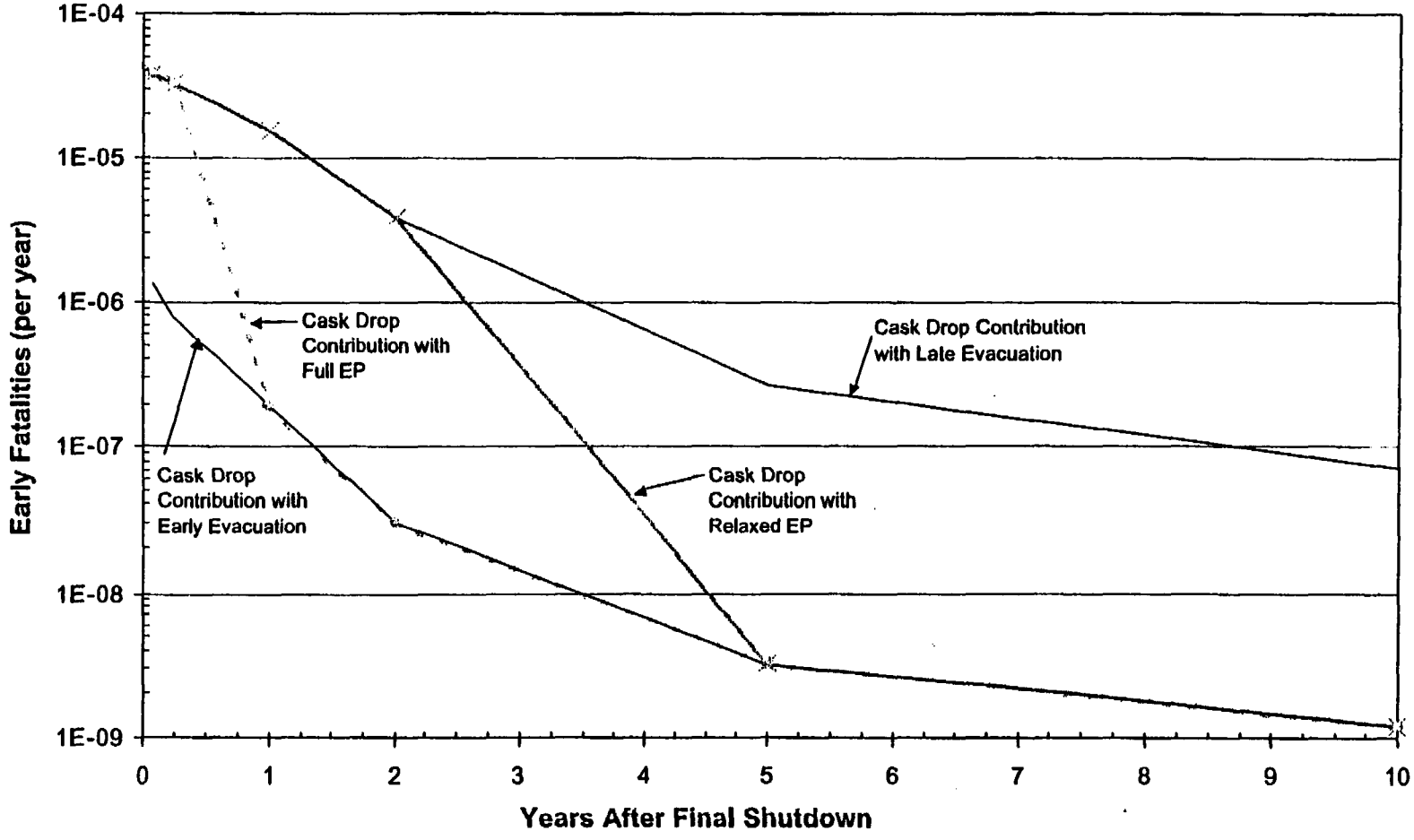

Figure 3.7-5

000172

Sensitivity of Societal (Person-rem) Risk to Emergency Planning -- Cask Drop Event

(Conditional upon: High Ruthenium Source Term)

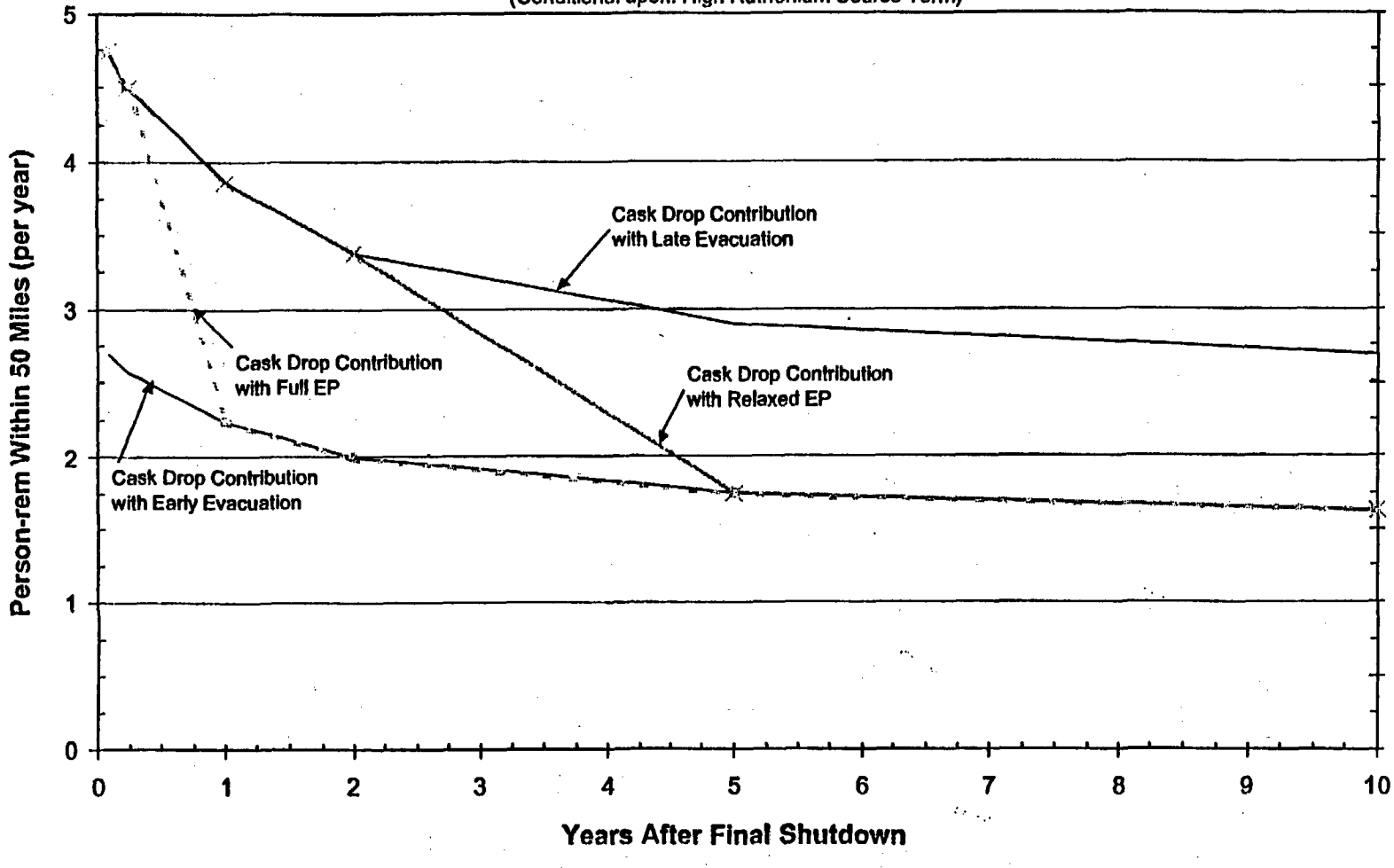

Figure 3.7-6

000173

On high ruthenium source term, The staff concludes:

- For the first 1 to 2 years following shutdown, the early fatality risk for an SFP fire is low, but may be comparable to that for a severe accident in an operating reactor (based on the two operating reactors considered). At 5 years following shutdown, the early fatality risk for SFP accidents is approximately two orders of magnitude lower than at shutdown. This is attributable to the effect of ruthenium, which decays to negligible amounts at 5 years.

- Societal risk for an SFP fire may be comparable to that for a severe accident in an operating reactor, but does not exhibit a substantial reduction with time because of the slower decay of fission products and the interdiction modeling assumptions that drive long-term doses.

- Of the SFP accidents assessed, only the cask drop accident is affected by changes to EP requirements. However, these changes do not substantially impact the total risk because the frequency of cask drop accidents is very low. As discussed previously, changes to EP requirements affect only the risk from cask drop accidents in the time period between 1 and 5 years.

- These observations are valid regardless of whether seismic event frequencies are based on the LLNL or the EPRI seismic hazard study.

About the low ruthenium source term the staff concludes:

- Use of the low ruthenium source term reduces early fatality risk by about a factor of 100 (relative to the high ruthenium source term) within the first 1 to 2 years and by about a factor of 10 at 5 years and after.

- With the low ruthenium source term, the early fatality risk for SFP accidents is about an order of magnitude lower than the corresponding values for a reactor accident shortly following shutdown and about two orders of magnitude lower at 2 years following shutdown. (In making these comparisons it is important to compare the SFP risks based on a particular seismic hazard estimate, e.g., EPRI, with reactor accident risks based on the same hazard estimate.)

- With the low ruthenium source term, the societal risk for SFP accidents is also about an order of magnitude lower than the corresponding values for a reactor accident shortly following shutdown, but does not exhibit a substantial reduction with time because of the slower decay of fission products and the interdiction modeling assumptions (discussed in Appendix 4) that drive long-term doses. Substantial reductions would only occur after about 5 years, when sufficient time appears to be available to initiate unplanned accident management recovery actions.

- As with the high ruthenium source term, changes to EP requirements affect the cask drop accident, and do not substantially impact the total risk due to the low frequency of cask drop accidents.

- These observations are valid regardless of whether seismic event frequencies are based on the LLNL or the EPRI seismic hazard estimates.

Figures 3.7-7 and 3.7-8 show the risk measures relevant to the Commission's safety goal policy statement, specifically, the individual risk of early fatality (to an individual within 1 mile of the site) and the individual risk of latent cancer fatality (to an individual within 10 miles of the site). The upper curves are based on the LLNL seismic hazard curves and the high ruthenium source term, and the lower curves are based on the EPRI hazard curves and the low ruthenium source term. Accordingly, these results may be viewed as a representative range of risk results for spent fuel pools uncovery given the conservative assumption that all SFP accidents result in a fire.

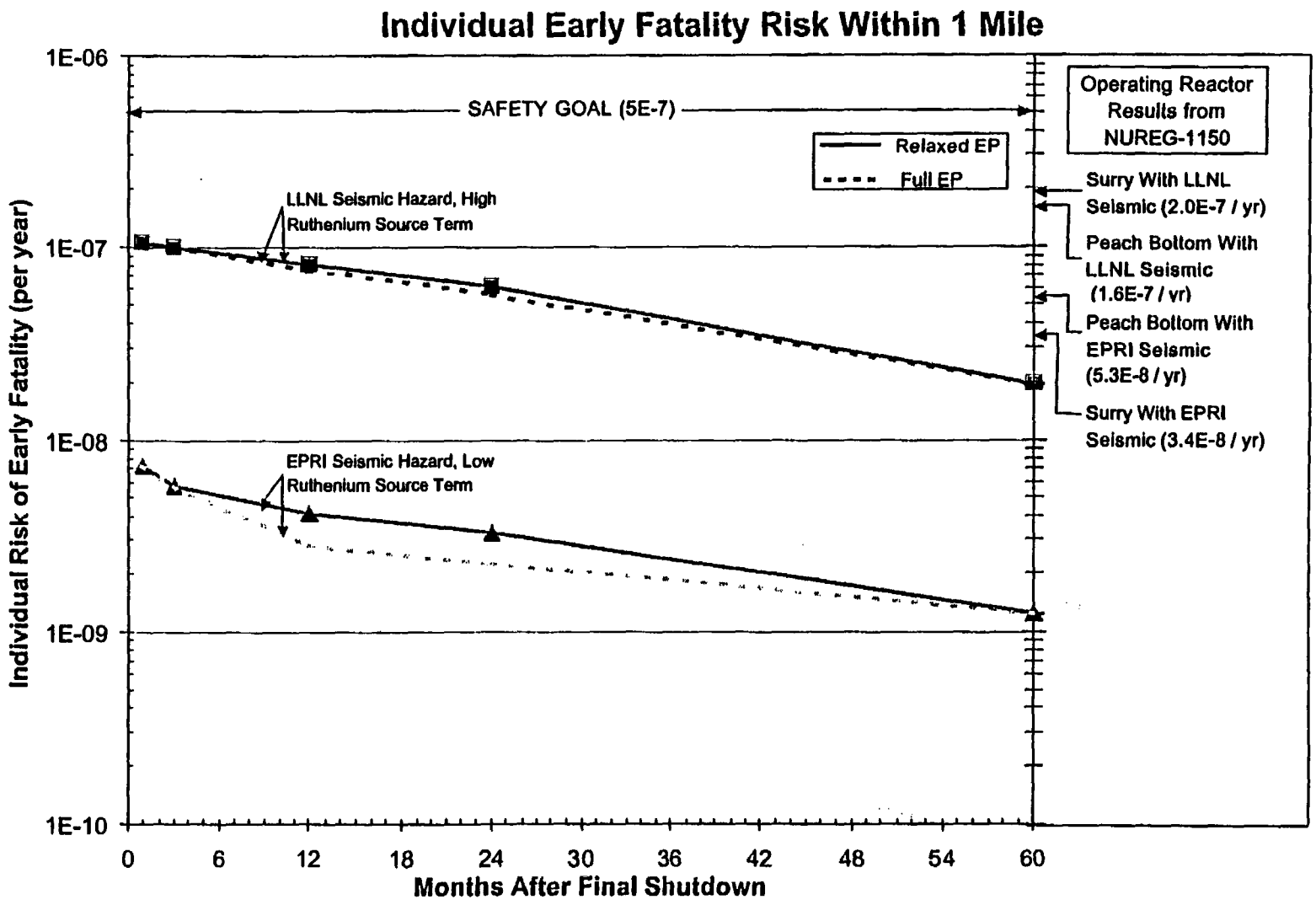

Figure 3.7-7

000176

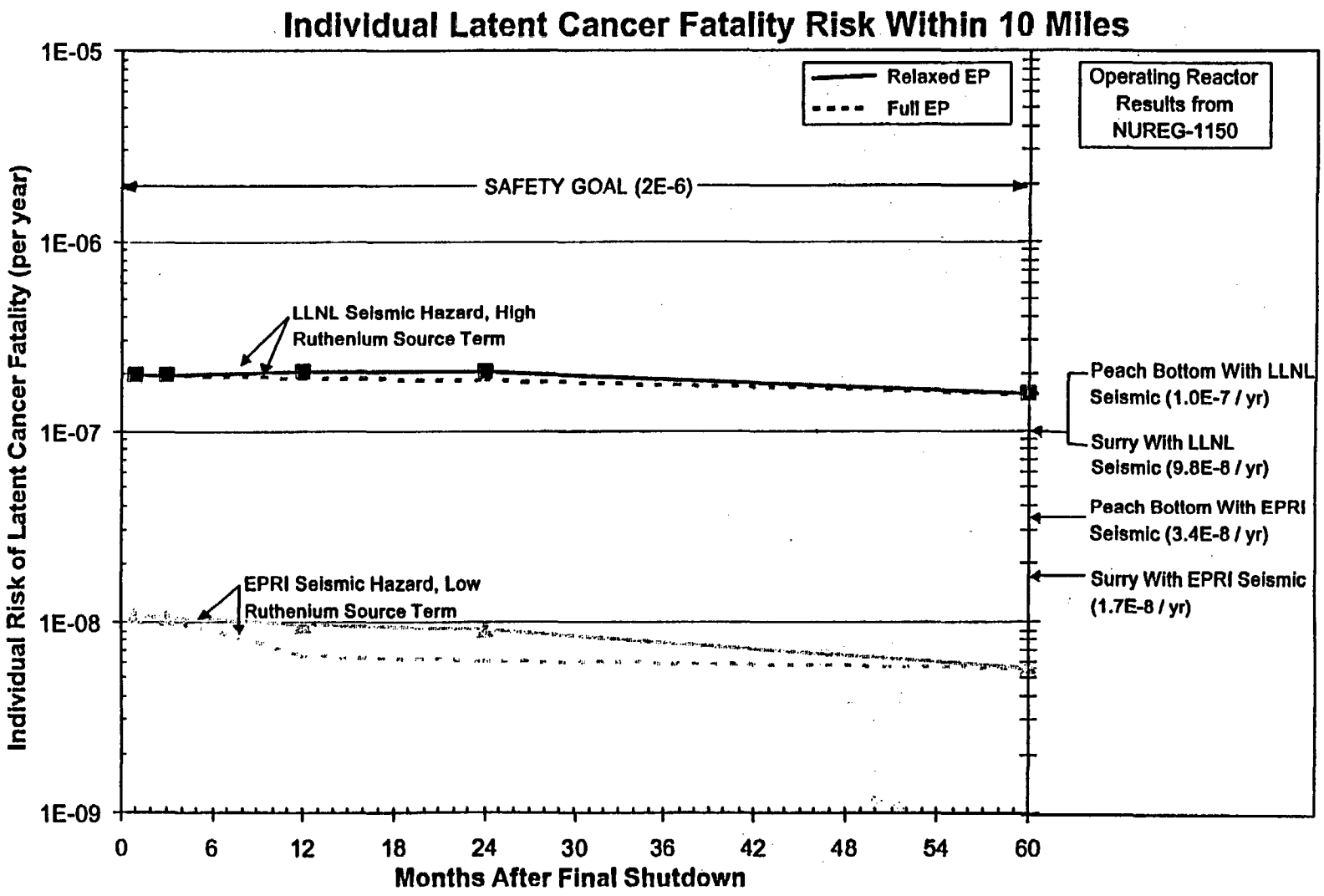

Figure 3.7-8

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The individual early fatality risk for an SFP accident is about one to two orders of magnitude lower than the Commission's safety goal, depending on assumptions about the SFP accident source term and seismic hazard. For the upper curve (corresponding to use of the mean of the LLNL seismic hazard estimates and the high ruthenium source term), the risks are about a decade lower than the safety goal. For the lower curve (corresponding to use of the mean of the EPRI seismic hazard estimates and the low ruthenium source term) the risks are about 2 decades lower than the safety goal. The individual early fatality risk for an SFP accident decreases with time and is about a factor of 5 lower at 5 years following shutdown (relative to the value at 30 days).

Similarly, the individual latent cancer fatality risk for an SFP accident is about one to two orders of magnitude lower than the Commission's safety goal, depending on assumptions about the SFP accident source term and seismic hazard. For the upper curve (corresponding to use of the mean of the LLNL seismic hazard estimates and the high ruthenium source term), the risks are about a decade lower than the safety goal. For the lower curve (corresponding to use of the mean of the EPRI seismic hazard estimates and the low ruthenium source term), the risks are about 2 decades lower than the safety goal. The individual latent cancer fatality risk for an SFP accident are not substantially reduced with time because of the slower decay of fission products and the interdiction modeling assumptions that drive long-term doses.

Changes to EP requirements, as modeled, do not substantially impact the margin between SFP risk and the safety goals because of the low frequency of events for which EP would be effective.

4.0 IMPLICATIONS OF SPENT FUEL POOL (SFP) RISK FOR REGULATORY REQUIREMENTS

The primary purpose of this study is to provide risk insights to support possible revisions to regulatory requirements for decommissioning plants. Section 4.1 below describes the safety principles of Regulatory Guide (RG) 1.174 as they apply to an SFP, and examines the design, operational, and regulatory elements that are important in ensuring that the risk from an SFP continues to meet these principles. This technical assessment explores possible implications for EP requirements, but the same technical information also provides risk insights to inform regulatory decisions on changes in insurance, safeguards, staffing and training, backfit, and other requirements for decommissioning plants. Section 4.2 examines the implications of the technical results for these regulatory decisions, and how future regulatory activity might reflect commitments and assumptions. The implications of safeguards events are not included in this evaluation.

4.1 Risk-Informed Decision Making

In 1995, the NRC published its PRA policy statement (Ref. 1), which stated that the use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art of the methods. Subsequent to issuance of the PRA policy statement, the agency published RG 1.174, which contained general guidance for application of PRA insights to the regulation of nuclear reactors. The regulatory framework proposed in this study for decommissioning plants is based on the risk-informed decision-making process described in RG 1.174 (Ref. 2). Although the focus of RG 1.174 is decision making regarding changes to the licensing basis of an operating plant, the same risk-informed philosophy can be applied generically as part of the evaluation of potential exemptions or changes to current regulatory requirements for decommissioning plants.

RG 1.174 articulates the following safety principles, which can be applied in evaluating regulatory changes for decommissioning plants:

- The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change, i.e., a "specific exemption" under 10 CFR 50.12 or a "petition for rulemaking" under 10 CFR 2.802.
- When proposed changes result in an increase in core damage frequency and/or risk, the increases should be small and consistent with the intent of the Commission's safety goal policy statement.
- The proposed change is consistent with the defense-in-depth philosophy.
- The proposed change maintains sufficient safety margins.
- The impact of the proposed change should be monitored using performance measurement strategies.

A discussion of each of these safety principles and how they would continue to be satisfied at a decommissioning plant is provided in the sections that follow. Since the application of this study specifically relates to exemptions to a rule or a rule change for decommissioning plants, a discussion of the first principle regarding current regulations is not necessary nor is it provided.

4.1.1 Increases in Risk

RG 1.174 states that when proposed changes result in an increase in core damage frequency and/or risk, the increases should be small and consistent with the intent of the Commission's safety goal policy statement.

The staff has evaluated the risks associated with SFP accidents and the impacts of potential changes to regulatory requirements for decommissioning plants relative to applicable regulatory guidance. Guidance on acceptable levels of (total) risk to the public from nuclear power plant operation is provided in the Commission's safety goal policy statement (Ref. 3). Additional guidance on the acceptable levels of risk increase from a change to the plant licensing basis is provided in RG 1.174. The guidance contained in these documents is summarized below and used in this study to evaluate the risks associated with SFP accidents and the impacts of potential changes to regulatory requirements for decommissioning plants.

SFP Risk Relative to the Safety Goal Policy Statement

The "Policy Statement on Safety Goals for the Operation of Nuclear Power Plants," issued in 1986, establishes goals that broadly define an acceptable level of radiological risk to the public as a result of nuclear power plant operation. These goals are used generically to assess the adequacy of current requirements and potential changes to the requirements. The Commission established two qualitative safety goals that are supported by two quantitative objectives for use in the regulatory decision-making process. The qualitative safety goals stipulate the following:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

- Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The following quantitative health objectives (QHOs) are used in determining achievement of the safety goals:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of 1 percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of 1 percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

These QHOs have been translated into two numerical objectives as follows:

- The individual risk of a prompt fatality from all "other accidents to which members of the U.S. population are generally exposed," such as fatal automobile accidents, is about $5x10^{-4}$ per year. One-tenth of 1 percent of this figure implies that the individual risk of prompt fatality from a reactor accident should be less than $5x10^{-7}$ per reactor year.

- "The sum of cancer fatality risks resulting from all other causes" for an individual is taken to be the cancer fatality rate in the U.S. which is about 1 in 500 or $2x10^{-3}$ per year. One-tenth of 1 percent of this risk means that the risk of cancer to the population in the area near a nuclear power plant due to its operation should be limited to $2x10^{-6}$ per reactor year.

Although the policy statement and related numerical objectives were developed to address the risk associated with power operation, the QHOs provide a convenient benchmark for SFP evaluations. Accordingly, the staff has compared the estimated risks associated with SFP accidents to the QHOs.

The risks associated with SFP accidents compare favorably with the QHOs. The comparisons, presented in Section 3.7.3, show that a typical site that conforms with the IDCs and SDAs would meet the QHOs by one to two orders of magnitude a few months following shutdown and by greater margins later. The risk comparisons provided in Appendix 4C show that SFP facilities maintained at or below the recommended pool performance guideline (PPG) of $1X10^{-5}$ per year, would continue to meet the QHO even with a severe SFP source term. With the exception of H.B. Robinson (using the LLNL seismic hazard estimates and generic fragilities), all Central and Eastern U.S. plants which satisfy the IDCs and SDAs (and pass the seismic checklist) will meet the PPG. Western plants (including San Onofre, Diablo Canyon, and WNP-2) were not included in the LLNL or EPRI seismic hazard studies and need to demonstrate compliance with the PPG on a plant-specific basis.

Risk Increases Relative to Regulatory Guide 1.174

The guidelines in RG 1.174 pertain to the core damage frequency (CDF) and large early release frequency (LERF). For both CDF and LERF, RG 1.174 contains guidance on acceptable values for the changes that can be allowed due to regulatory decisions as a function of the baseline frequencies. For example, if the baseline CDF for a plant is below $1x10^{-4}$ per year, plant changes can be approved that increase CDF by up to $1x10^{-5}$ per year. If the baseline LERF is less than $1x10^{-5}$ per year, plant changes can be approved which increase LERF by up to $1x10^{-6}$ per year.

For decommissioning plants, the risk is primarily due to the possibility of a zirconium fire involving the spent fuel cladding. The consequences of such an event do not equate directly to either a core damage accident or a large early release as modeled for an operating reactor. Zirconium fires in SFPs have the potential for significant long-term consequences because

multiple cores may be involved; the relevant cladding and fuel degradation mechanisms could lead to increased releases of certain isotopes (e.g., short-lived isotopes such as iodine will have decayed, but the release of long-lived isotopes such as ruthenium could be increased due to air-fuel reactions); and there is no containment surrounding the SFP to mitigate the consequences. On the other hand, after about 2 years, the consequences are different than from a large early release because the postulated accidents progress more slowly, allowing time for protective actions to be taken to significantly reduce early fatalities (and to a lesser extent latent fatalities). In effect, an SFP fire would result in a "large" release, but this release would not generally be considered "early" due to the significant time delay before fission products are released.

In spite of the differences relative to an operating reactor large early release event and the differences in isotopic makeup, the consequence calculations performed by the staff and discussed in Section 3.7 show that SFP fires could have health effects comparable to those of a severe reactor accident. These calculations considered the effects of different source terms and evacuation assumptions on offsite consequences. Since an SFP fire scenario would involve a direct release to the environment with significant consequences, the staff has decided that the RG 1.174 guidance concerning LERF can be applied to the issue of SFP risks for decommissioning plants.

The LERF guidance is applied in two ways in this study:

(1) Because the changes in EP requirements affect not the frequency of events involving a large early release (i.e., the SFP fire frequency) but the consequences of these releases, the allowable increase in LERF in RG 1.174 is translated into an allowable increase in key risk measures. The estimated risk increases associated with changes in EP requirements are then compared to the allowable increases inferred from RG 1.174. These comparisons are presented in Appendix 4D.

(2) The RG 1.174 guidance is used to establish a PPG. The PPG provides a threshold for controlling the risk from a decommissioning SFP. By maintaining the frequency of events leading to uncovery of the spent fuel at a value less than the recommended PPG value of 1×10^{-5} per year, zirconium fires will remain highly unlikely, the risk will continue to meet the Commission's QHOs, and changes to the plant SFP licensing basis that result in very small increase in risk may be permitted consistent with the logic in RG 1.174. A licensee would need to assure that the frequency of events leading to uncovery of the spent fuel would be less than the PPG in order to consider the risk-informed changes in a rule for decommissioning plants. With the exception of those plants mentioned above, this assurance could be provided by conforming with the IDCs and SDAs listed in Tables 4.2-1 and -2. The use of the LERF guidance (1×10^{-5} per year frequency of fuel uncovery) was questioned by the ACRS because of concerns related to SFP source terms and accident consequences. The rationale for the PPG is presented in Appendix 4C.

The risk increases associated with relaxations in EP requirements compare favorably with the guidance contained in RG 1.174 (see Table 4 of Appendix 4D). Relaxation of EP requirements would result in an increase of about 1.5×10^{-5} early fatalities per year and 2 person-rem per year for the Surry analysis, the first is about a factor of 15 and the second a factor of 5 below the allowable increase inferred from the RG 1.174 LERF criteria. The increase in the QHO risk measures is also substantially lower than that allowed in RG 1.174. Since the SFP fire

frequency assumed in these comparisons is about a factor of 4 lower than the PPG of 1×10^{-5} per year, an SFP facility operating nominally at the PPG would have a smaller margin to the allowable risk limits for the reference plant but would still be at or below these limits.

As discussed in Section 3.7, the basis for these results is that EP is of marginal benefit in large earthquakes because of offsite damage. However, as described in Appendix 4D, even with unrealistically optimistic assumptions about the effectiveness of EP in seismic events (i.e., assuming full and relaxed EP results in early and late evacuation, respectively, and using the LLNL seismic hazard frequency and the high ruthenium source term), the change in risk is small and the QHOs continue to be met with adequate margin.

Measures to Assure Risk Increases Remain Small

The analysis in Section 3 explicitly examines the risk impact of specific design and operational characteristics. This analysis credits the industry decommissioning commitments (IDCs) proposed by NEI in a letter to the NRC dated November 12, 1999 (Ref. 4) and several additional staff decommissioning assumptions (SDAs) identified through the staff's risk assessment and the staff's evaluation of the RG 1.174 safety principles for decommissioning plants. The IDCs and SDAs are summarized in Tables 4.2-1 and 4.2-2.

The low numerical risk results shown in Section 3 and Appendix 2 are predicated on the IDCs and SDAs being fulfilled. Specifically,

- IDC #5 and SDAs #2 and #3 provide assurance of timely operator response for a broad range of operational events.
- The low likelihood of pool failure due to heavy load drop is dependent on design and procedural controls for handling of heavy loads (IDC #1 and #9 and SDA #5).
- The low baseline frequency for a seismically initiated zirconium fire is predicated upon implementation of the seismic checklist shown in Appendix 5 (SDA #6).
- The low likelihood of loss of cooling is dependent upon procedures and training (IDC #2) and instrumentation (IDC #5 and SDA #3).
- The low likelihood of loss of inventory is dependent upon design provisions (IDC #6) and procedures and controls (IDC #7) to limit leakage.
- The high probability that the operators will identify and recover from a loss of cooling or a loss of inventory event is dependent upon procedures and training for effective use of onsite and offsite resources (IDCs #2 through #4, IDC #8, and SDA #3) and SFP instrumentation (IDC #5 and SDA #3).
- The low likelihood criticality issues is dependent on continuation of programs to assess the condition of Boraflex absorber material (SDA #7).
- Applicability of the staff's generic risk assessment to a specific facility is assured by SDA #1.

With regard to SFP risks and risk increases associated with EP relaxations, the staff concludes:

- An SFP facility that conforms with the IDCs and SDAs would meet the QHOs by one to two orders of magnitude shortly after shutdown and with greater margins at later times.

- The risk increase associated with relaxations in EP requirements is very small, even under assumptions that maximize the effectiveness of emergency preparedness in seismic events (i.e., assigning consequences for the "full EP" case based on early evacuation and consequences for the "relaxed EP" case based on late evacuation), and the QHOs continue to be met with adequate margin.

- Continued conformance with IDCs and SDAs provides reasonable assurance that the SFP risk and risk increases associated with regulatory changes would remain small.

4.1.2 Defense-in-Depth

RG 1.174 states that the proposed change should be consistent with the defense-in-depth philosophy.

In accordance with the Commission white paper on risk-informed regulation (March 11, 1999), "Defense-in-depth is an element of the NRC's Safety Philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. The defense-in-depth philosophy ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of a nuclear facility. The net effect of incorporating defense-in-depth into design, construction, maintenance and operation is that the facility or system in question tends to be more tolerant of failures and external challenges." Therefore, application of defense-in-depth could mean in part that there is more than one source of cooling water or that pump make-up can be provided by both electric as well as direct-drive diesel pumps. Additionally, defense-in-depth can mean that even if a serious outcome (such as fuel damage) occurs, there is further protection, such as containment for operating plants to prevent radionuclide releases to the environment and emergency response measures to provide dose savings to the public.

The defense-in-depth philosophy applies to the operation of the SFP in a decommissioning plant. The philosophy also applies to the potential regulatory changes contemplated for decommissioning plants. Implementation of defense-in-depth for SFPs is different than for nuclear reactors because the hazards are different. The robust structural design of a fuel pool, coupled with the simple nature of the pool support systems, goes far toward preventing accidents associated with loss of water inventory or pool heat removal. Additionally, because the essentially quiescent (low-temperature, low-pressure) initial state of the SFP and the long time available for taking corrective action associated with most release scenarios provide significant safety margin, a containment structure is not considered necessary as an additional barrier to provide an adequate level of protection to the public. Likewise, the slow evolution of most SFP accident scenarios allows for reasonable human recovery actions to respond to system failures, and provides sufficient time to allow for the implementation of protective actions.

The staff's risk assessment demonstrates that the risk from a decommissioning plant SFP accident is small if IDCs and additional SDAs are implemented as assumed in the risk study. Due to the different nature of an SFP accident versus an accident in an operating reactor with respect to system design capability and event timing, the defense-in-depth function of reactor containment is not required. However, the staff has found that defense-in-depth in the form of accident prevention measures and an appropriate level of emergency planning can limit risk and provide dose savings for as long as a zirconium fire is possible.

Defense-in-depth for accident prevention and mitigation is provided by licensee conformance with the IDCs and SDAs, as discussed previously. Defense-in-depth for consequence mitigation should continue to be provided by retaining requirements for an appropriate level of EP in consideration of the amount of time available before fission product release in specific events.

For the purpose of the analysis in this study, when referring to relaxation of offsite EP, the study assumed conditions that would be similar to those at sites in decommissioning that have already received exemptions from some EP requirements. For instance, licensees may no longer be required: to have a formalized emergency planning zone (EPZ); to have an emergency operations facility (EOF), technical support center (TSC), or operations support center (OSC); to promptly notify the public using a siren system, tone alert radios, or National Weather Service radios; or to conduct biennial full-participation exercises. The analyses in the study were simplified to focus on conditions which assumed evacuation occurred either early or late.

It is understood that EP involves more than just evacuation considerations. In the analysis of the study, it was assumed that the decommissioning licensee would still be required to notify offsite authorities, characterize the releases, and make protective action recommendations; have a means of notifying offsite organizations and providing information to the public; and hold onsite biennial exercises and semiannual drills.

The assessments conducted for this study show that, 60 days after final shutdown, recovery and mitigation times of more than 100 hours are available before release occurs, except for the most severe events. These times appear to be sufficient to permit offsite protective actions to be implemented on an ad hoc basis, if necessary, without the full compliment of regulatory requirements associated with operating reactors. The staff notes that potential relaxation of EP requirements for decommissioning plants could be phased in such that the relaxation would not result in an immediate lapse of all offsite emergency response capabilities following final shutdown, but would more likely result in early elimination of some capabilities (e.g., sirens) and more gradual relaxation of certain other capabilities (e.g., pre-planning of evacuations and communications), with a transition towards longer ad hoc response times over several years due to such factors as attrition of experienced personnel. Shortly after final shutdown, when SFP heatup rates and risks are greatest, response capabilities are expected to be largely intact and comparable to those for full EP. These capabilities could be expected to diminish over time, resulting in longer ad hoc response times. However, continued fission product decay in the spent fuel will result in longer times to release, providing additional time during which emergency response measures could be implemented.

Only during the first several years and in the most severe events, such as severe seismic events, heavy load drops, and other dynamic events, that cause the pool to fail, would the accident progress so rapidly that emergency response measures might not be implemented in a

timely manner. The staff's risk study indicates that the frequency of such events is dominated by earthquakes with a magnitude several times the safe shutdown earthquake (SSE). As discussed in Section 3.7.2, for ground motion levels that correspond to SFP failure, emergency planning would have marginal benefit because of extensive collateral damage to infrastructure (e.g., power, communications, buildings, roads, and bridges). Emergency response action would likely require substantial ad hoc action regardless of pre-planned actions in these events.

The next largest contribution is from cask drop sequences. The frequency of such events is low in the staff's risk study ($2x10^{-7}$ per year) due to implementation of IDCs and SDAs concerning movement of heavy loads. Relaxations in EP requirements could result in some increase in the risk associated with these events for a limited time following shutdown (1 to 5 years in the staff's analysis). However, the increase is a small fraction of the total risk from SFPs, as shown in Section 3.7. For the remaining SFP accidents that were analyzed and lead to SFP fires (e.g., boildown sequences due to organizational failures), current emergency planning was assumed to be ineffective or the frequencies of accidents, (e.g., aircraft impact) would be at least an order of magnitude lower than for the cask drop accident. Thus, mitigation of these events would not be risk significant.

With regard to defense-in-depth, the staff concludes:

- Defense-in-depth for accident prevention and mitigation is provided by the robust design of the SFP, the simple nature of pool support systems, and the long time available for taking corrective action in response to system failures.

- The substantial amount of time available for ad hoc offsite emergency response should provide some level of defense-in-depth for consequence mitigation in SFP accidents.

- In the large seismic events that dominate SFP risk, pre-planning for radiological accidents would have marginal benefit due to extensive collateral damage offsite. Accordingly, relaxations in EP requirements are not expected to substantially alter the outcome from such a large seismic event.

- There can be a tradeoff between the formality with which the elements of emergency planning (procedures, training, performance of exercises) are treated and the increasing safety margin as the fuel ages and the time available to respond gets longer.

4.1.3 Safety Margins

RG 1.174 states that the proposed change should maintain sufficient safety margins.

As discussed in Section 2, the safety margins associated with fuel in the SFP are much greater than those associated with an operating reactor due to the low heat removal requirements and long time frames available for recovery from off-normal events. Due to these larger margins, the staff judges that the skid-mounted and other dedicated SFP cooling and inventory systems in place provide adequate margins for accident prevention. Additionally, the presence of soluble boron or Boraflex provides additional assurance of margin with respect to shutdown reactivity.

The risk results provided in Section 3.6.3 show that a typical site that conforms with the IDCs and SDAs would meet the Commission's QHOs by one to two orders of magnitude, depending on assumptions about the SFP source term and seismic hazard frequency. The risk comparisons provided in Appendix 4C show that SFP facilities maintained at or below the recommended PPG of $1x10^{-5}$ per year would continue to meet the QHOs for even the most severe source term.

The estimated risk increases associated with the EP relaxations are also well below the allowable increases developed from the RG 1.174 LERF criteria. As discussed in Section 4.2.1 and Appendix 4D, the increases in risk from the EP relaxation would be about a factor of 10 below the maximum allowable increases developed from RG 1.174. Since the SFP fire frequency assumed in the RG 1.174 comparisons is about a factor of 4 lower than the PPG of $1x10^{-5}$ per year, a plant operating nominally at the PPG would have a smaller margin to the allowable risk limits for the reference plant but would still be at or below the limits.

The results of a sensitivity case in Appendix 4D indicate that even with assumptions that maximize the effectiveness of EP in seismic events, the change in risk associated with relaxation of the requirements for radiological preplanning is still relatively small. The increases in early fatalities and individual early fatality risk remain below the maximum allowable for each risk measure. Population dose and individual latent cancer fatality risk are about a factor of 2 higher than the allowable value inferred from RG 1.174. This increase in individual latent cancer risk represents about 9 percent of the QHO; thus, considerable margin to the QHO would still remain.

The evacuation effectiveness assumed for "full EP" in the sensitivity case is unrealistic for high ground motion earthquakes, and the risk increase associated with the EP relaxations is expected to be closer to the baseline value. Also, the risk reduction estimates are based on the LLNL seismic hazard frequencies and the high ruthenium source term, and would be substantially lower if either the EPRI seismic hazard frequencies or the low ruthenium source term were used. The above comparisons are based on the risk levels 1 year after shutdown but would also be valid several months following shutdown. Use of either the EPRI seismic hazard frequencies or the low ruthenium source term would reduce each of the risk measures by about a factor of 10, to values well below the RG 1.174 guidelines and the QHOs. The risk impact will decrease even further in later years due to reduced consequences as fission products decay.

The study concludes that relaxation of certain EP requirements can be considered for decommissioning plants in which conformance with the IDCs and SDAs provides reasonable assurance that sufficient margins to the safety goals will be maintained.

4.1.4 Implementation and Monitoring Program

RG 1.174 states that the impact of the proposed change should be monitored using performance measurement strategies. RG 1.174 further states that an implementation and monitoring plan should be developed to ensure that the engineering evaluation conducted to examine the impact of the proposed changes continues to reflect the actual reliability and availability of SSCs that have been evaluated. This will ensure that the conclusions that have been drawn will remain valid.

Applying this guideline for the SFP risk evaluation results in identification of four primary areas for performance monitoring: (1) the performance and reliability of SFP cooling and associated power and inventory makeup systems, (2) the Boraflex condition for high-density fuel racks, (3) crane operation and load path control for cask movements, and (4) onsite emergency response capabilities. The following monitoring should continue after decommissioning in order to assure SFP risk remains low:

- Performance and reliability monitoring of the SFP systems, heat removal, AC power, and inventory should comply with the provisions of the Maintenance Rule (10 CFR 50.65).

- The current monitoring programs identified in licensee's responses to Generic Letter 96-04 (Ref. 2) with respect to monitoring of the Boraflex absorber material should be maintained by decommissioning plants until all fuel is removed from the SFP. This staff assumption is stated in SDA #7 (see Table 4.1-2).

- Heavy load activities and load paths should be monitored and controlled by the licensee in accordance with IDC #1 (see Table 4.1-1).

- Licensees should continue to provide a level of onsite capabilities to assure prompt notification of offsite authorities, characterization of potential releases, development of protective action recommendations, and communication with the public. These capabilities should be monitored by holding periodic onsite exercises and drills.

The staff concludes that continued compliance with the Maintenance Rule, the IDCs, and the SDAs, together with some level of EP, provides a reasonable level of monitoring of SFP safety.

Table 4.1-1 Industry Decommissioning Commitments (IDCs)

IDC No.	Industry commitments
1	Cask drop analyses will be performed or single failure-proof cranes will be in use for handling of heavy loads (i.e., phase II of NUREG-0612 will be implemented).
2	Procedures and training of personnel will be in place to ensure that onsite and offsite resources can be brought to bear during an event.
3	Procedures will be in place to establish communication between onsite and offsite organizations during severe weather and seismic events.
4	An offsite resource plan will be developed which will include access to portable pumps and emergency power to supplement onsite resources. The plan would principally identify organizations or suppliers where offsite resources could be obtained in a timely manner.
5	SFP instrumentation will include readouts and alarms in the control room (or where personnel are stationed) for SFP temperature, water level, and area radiation levels.
6	SFP seals that could cause leakage leading to fuel uncovery in the event of seal failure shall be self limiting to leakage or otherwise engineered so that drainage cannot occur.
7	Procedures or administrative controls to reduce the likelihood of rapid draindown events will include (1) prohibitions on the use of pumps that lack adequate siphon protection or (2) controls for pump suction and discharge points. The functionality of anti-siphon devices will be periodically verified.
8	An onsite restoration plan will be in place to provide repair of the SFP cooling systems or to provide access for makeup water to the SFP. The plan will provide for remote alignment of the makeup source to the SFP without requiring entry to the refuel floor.
9	Procedures will be in place to control SFP operations that have the potential to rapidly decrease SFP inventory. These administrative controls may require additional operations or management review, management physical presence for designated operations or administrative limitations such as restrictions on heavy load movements
10	Routine testing of the alternative fuel pool makeup system components will be performed and administrative controls for equipment out of service will be implemented to provide added assurance that the components would be available, if needed.

Table 4.1-2 Staff Decommissioning Assumptions (SDAs)

SDA No.	Staff Assumptions
1	Licensee's SFP cooling design will be at least as capable as that assumed in the risk assessment, including instrumentation. Licensees will have at least one motor-driven and one diesel-driven fire pump capable of delivering inventory to the SFP.
2	Walk-downs of SFP systems will be performed at least once per shift by the operators. Procedures will be developed for and employed by the operators to provide guidance on the capability and availability of onsite and offsite inventory makeup sources and time available to initiate these sources for various loss of cooling or inventory events.
3	Control room instrumentation that monitors SFP temperature and water level will directly measure the parameters involved. Level instrumentation will provide alarms at levels associated with calling in offsite resources and with declaring a general emergency.
4	Licensee determines that there are no drain paths in the SFP that could lower the pool level (by draining, suction, or pumping) more than 15 feet below the normal pool operating level and that licensee must initiate recovery using offsite sources.
5	Load Drop consequence analyses will be performed for facilities with non-single failure-proof systems. The analyses and any mitigative actions necessary to preclude catastrophic damage to the SFP that would lead to a rapid pool draining would be sufficient to demonstrate that there is high confidence in the facilities ability to withstand a heavy load drop.
6	Each decommissioning plant will successfully complete the seismic checklist provided in Appendix2B to this study. If the checklist cannot be successfully completed, the decommissioning plant will perform a plant specific seismic risk assessment of the SFP and demonstrate that SFP seismically induced structural failure and rapid loss of inventory is less than the generic bounding estimates provided in this study ($<1x10^{-5}$ per year including non-seismic events).
7	Licensees will maintain a program to provide surveillance and monitoring of Boraflex in high-density spent fuel racks until such time as spent fuel is no longer stored in these high-density racks.

4.2 Implications for Regulatory Requirements for Emergency Preparedness, Security, and Insurance

The industry and other stakeholders have expressed interest in knowing the relevance of the results of this study to decisions on specific regulatory requirements. These decisions could be made in response to plant-specific exemption requests or as part of the integrated rulemaking for decommissioning plants. Such decisions can be facilitated by a risk-informed examination of both the deterministic and probabilistic aspects of decommissioning. Three examples of such regulatory decisions are presented in this section: regulatory requirements for emergency preparedness, security, and insurance.

4.2.1 Emergency Preparedness

The requirements for emergency preparedness are contained in 10 CFR 50.47 (Ref. 5) and Appendix E to 10 CFR Part 50 (Ref. 6). Further guidance on the basis for EP requirements is contained in NUREG-0396 (Ref. 7) and NUREG-0654/FEMA-REP-1 (Ref. 8). The task force of NRC and Environmental Protection Agency (EPA) representatives formed to address the planning bases for emergency preparedness concluded that the overall objective of EP is to provide dose savings (and in some cases immediate life saving) for a spectrum of accidents that could produce offsite doses in excess of predetermined protective action guides (PAGs).

In the past, the NRC staff has typically granted exemptions from offsite emergency planning requirements for decommissioning plants that could demonstrate that they were beyond the period in which a zirconium fire could occur. The rationale for those decisions was that, in the absence of a zirconium fire, there were no decommissioning plant scenarios for which the consequences justify the imposition of an offsite EP requirement. The results of this technical study confirm that the frequency of events leading to SFP fires is very low (ranging from about 4×10^{-7} at sites where seismic events are a minimal contributor to less than 1×10^{-5} per year at sites where seismic events dominate SFP risk and no plant-specific seismic analyses need to be performed), and that the subset of events in which EP can produce significant dose savings is even smaller (about 2×10^{-7} per year). However, the staff concludes that the possibility an SFP accident will lead to a large fission product release cannot be ruled out even many years after final shutdown, since several SFP accidents could involve either blockage of the air cooling path (e.g., due to partial draining of the SFP) or inadequate air circulation within the SFP building, resulting in near-adiabatic heatup of the spent fuel. The impact of this new information on previously granted exemptions is being evaluated by the staff. Large seismic events that fail the SFP are the dominant contributor to these failure modes. Emergency planning would be of marginal benefit in reducing the risk of such events due to its impairment by offsite damage. The next largest contributor, cask drop accidents, is about an order of magnitude lower in frequency. In the first few years following final shutdown (when time to fission product release is less than about 10 hours), EP could provide some dose savings, but does not substantially impact risk due to the low frequency of these events. Finally, although large releases from the SFP would remain possible for these failure modes, the time available before release would be in excess of 24 hours 5 years after final shutdown and sufficient to support implementation of protective measures on an ad hoc basis.

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In some cases, emergency preparedness exemptions have also been granted to plants which were still in the window of vulnerability for zirconium fire. In these cases, the justification was that enough time had elapsed since shutdown so that the evolution of a zirconium fire accident would evolve slowly enough to allow mitigative measures and, if necessary, offsite protective actions to be implemented without preplanning. The staff believes that the technical analysis discussed in Section 3 and the decision criteria laid out in Section 4.1 provides information on how such exemption requests could be viewed in the future. In addition, this information bears on the need for, and the extent of, emergency preparedness requirements in the integrated rulemaking. In consideration of the study's conclusion that air cooling may not always be available for some event sequences, the basis for some previous exemptions may need to be reconsidered.

The consequence analysis presented in Appendix 4 indicates that the offsite consequences of a zirconium fire may be comparable to those from operating reactor postulated severe accidents. Further, the analysis indicates that timely evacuation, implemented through either pre-planned or ad hoc measures, can significantly reduce the number of early fatalities due to a zirconium fire. The results in Section 3.7.3 indicate that early fatality and societal risk for an SFP fire may be comparable to that for an operating reactor, and that the risk is one to two orders of magnitude lower than the Commission's safety goal. The results in Appendix 4D show that even with the most optimistic assumptions about the effectiveness of EP in large seismic events, the increase in risk associated with relaxations in EP requirements is small and the QHOs continue to be met. Thus, the risk assessment provides some basis for reductions in EP requirements for decommissioning plants. With respect to the potential for pool criticality, the staff's assessment discussed in Section 3 and Appendix 3 indicates that credible scenarios for criticality are highly unlikely and are further precluded by the assumption of Boraflex monitoring programs. Additionally, even if a criticality event did occur, it would not have offsite consequences. Therefore, the conclusions regarding possible reductions in EP program requirements are not affected.

In Section 4.1, the safety principles of RG 1.174 are applied to assess whether changes to emergency preparedness requirements are appropriate. Notwithstanding the low risk associated with SFP accidents, the safety principles in RG 1.174 dictate that defense-in-depth be considered. As discussed previously, emergency preparedness provides defense-in-depth. However, because of the considerable time available to initiate protective actions, in most SFP accidents (and the low frequency of events in which sufficient time is not available to implement protective actions on an ad hoc basis), the level of formal emergency plans needed for rapid initiation and implementation of offsite protective actions can be evaluated. The principle emergency planning measures needed for SFP events is the means for identifying the event and notifying of State and local emergency response officials.

4.2.2 Security

Currently licensees that have permanently shut down reactor operations and have off-loaded the spent fuel into the SFP are still required to meet all the security requirements for operating reactors in 10 CFR 73.55 (Ref. 9). This level of security requires a site with a permanently shutdown reactor to provide security protection at the same level as for an operating reactor site. The industry has asked the NRC to consider whether the risk of radiological release from

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decommissioning plants due to sabotage is low enough to justify modification of safeguards requirements for SFPs at decommissioning plants.

In the past, decommissioning licensees have requested exemptions from specific regulations in 10 CFR 73.55 on the basis of the reduced number of target sets susceptible to sabotage attacks and the consequent reduced hazard to public health and safety. Limited exemptions have been granted on this basis. The risk analysis in this study does not refute the reduced target set argument; however, the analysis does not support the assertion of a lesser hazard to public health and safety, given the possible consequences of sabotage-induced uncovery of fuel in the SFP when a zirconium fire potential exists. Further, the risk analysis in this study did not evaluate the potential consequences of a sabotage event that could directly cause offsite fission product dispersion, for example, a vehicle bomb driven into or otherwise significantly damaging the SFP, even after a zirconium fire was no longer possible. However, this study supports a regulatory framework that relieves licensees from selected requirements in 10 CFR 73.55 on the basis of target set reduction when all fuel has been placed in the SFP.

As a result of the conclusions from this study, the bases for previous exemptions for defueled facilities, the devitalization of the spent fuel pool at operating reactors, and certain concerns at ISFSIs may need to be reconsidered. This is due to differences in the findings relative to the specific periods of time historically used for the devitalization of spent fuel pools at operating reactors and certain operational concerns and potential vulnerabilities at decommissioning sites.

The risk estimates contained in this study are based on accidents initiated by random equipment failures, human errors, or external events. PRA practitioners have developed and used dependable methods for estimating the frequency of such random events. By contrast, this analysis and PRA analyses in general do not include events due to sabotage. No established method exists for estimating the likelihood of a sabotage event. Nor is there a method for analyzing the effect of security provisions on that likelihood. Security regulations are designed and structured to prevent sabotage on the assumption that the design-basis threat could occur at commercial nuclear power plants without assessing the actual probability or consequences.

The technical information contained in this study shows that the consequences of a zirconium fire would be high. Moreover, the risk analysis could be used effectively to help determining priorities for, and details of, the security capability at a plant. However, no information in the analysis bears on the level of security necessary to limit the risk from sabotage events. Those decisions will continue to be made by a analytical assessment of the level of threat and the difficulty of protecting a specific facility.

4.2.3 Insurance

In accordance with 10 CFR Part 140 (Ref. 10), each 10 CFR Part 50 licensee is required to maintain public liability coverage in the form of primary and secondary financial protection. This coverage is required to be in place from the time unirradiated fuel is brought onto the facility site until all of the radioactive material has been removed form the site, unless the Commission terminates the Part 50 license or otherwise modifies the financial protection requirements under Part 140. On March 17, 1999, the staff proposed to the Commission that insurance indemnity requirements for permanently shutdown reactors be developed in an integrated, risk-informed effort along with emergency preparedness and security requirements. In the past, licensees

have been granted exemptions from financial protection requirements on the basis of deterministic analyses that indicate that a zirconium fire could no longer occur.

In the staff requirements memorandum (SRM) for SECY-93-127 (Ref. 11), the Commission suggested that withdrawal of secondary financial protection insurance coverage be allowed after the requisite minimum spent fuel cooling period had elapsed. Further, the Commission directed the staff to determine more precisely the appropriate spent fuel cooling period after plant shutdown. While insurance does not lend itself to a "small change in risk" analysis because insurance affects neither the probability nor the consequences of an event, the NRC staff has considered whether the risk analysis in this study justifies relief from this requirement for a decommissioning plant while it is vulnerable to zirconium fires. The risk analysis in the February 2000 study identified a generic window of vulnerability for an SFP fire until about 5 years after shutdown. The analysis in this study, however, indicates that a zirconium fire cannot be precluded on a generic basis even after 5 years decay. This is because a spent fuel configuration necessary to assure air cooling cannot be assured following a severe earthquake or cask drop event that drains the pool. Since a criteria of "sufficient cooling to preclude a fire" cannot be met and the long-term consequences could be significant (e.g., the long-term consequences (and risk) decrease very slowly because cesium-137 has a half life of approximately 30 years), the staff will need to consider alternative criteria if changes to insurance requirements are to be pursued.

5.0 SUMMARY AND CONCLUSIONS

This study documents an evaluation of spent fuel pool (SFP) accident risk at decommissioning plants. The study was undertaken to develop a risk-informed technical basis for reviewing exemption requests and a regulatory framework for integrated rulemaking. The staff tried to actively involve the public and industry representatives throughout the study. The staff held a series of public meetings with stakeholders during and after the preparation of a preliminary study (published in June 1999 at the request of the Nuclear Energy Institute (NEI)).

The staff published a draft study in February 2000 for public comment and significant comments were received from the public and the Advisory Committee on Reactor Safeguards (ACRS). To address these comments the staff did further analyses and also added sensitivity studies on evacuation timing to assess the risk significance of relaxed offsite emergency preparedness requirements during decommissioning. The staff based its sensitivity assessment on the guidance in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis." The staff's analyses and conclusions apply to decommissioning facilities with SFPs that meet the design and operational characteristics assumed in the risk analysis. These characteristics are identified in the study as industry decommissioning commitments (IDCs) and staff decommissioning assumptions (SDAs). Provisions for confirmation of these characteristics would need to be an integral part of rulemaking.

The results of the study indicate that the risk at SFPs is low and well within the Commission's Quantitative Health Objectives (QHOs). The risk is low because of the very low likelihood of a zirconium fire even though the consequences from a zirconium fire could be serious. Because of the importance of seismic events in the analysis, and the considerable uncertainty in seismic hazard estimates, the results are presented for both the Lawrence Livermore National Laboratory (LLNL) and the Electric Power Research Institute (EPRI) seismic hazard estimates. In addition, to address a concern raised by the ACRS, the results also include a sensitivity to a large ruthenium and fuel fines release fraction. The results indicate that the risk is well below the QHOs for both the individual risk of early fatality and the individual risk of latent cancer fatality.

The study includes use of a pool performance guideline (PPG) as an indicator of low risk at decommissioning facilities. The recommended PPG value for events leading to uncovery of the spent fuel was based on similarities in the consequences from a SFP zirconium fire to the consequences from a large early release event at an operating reactor. A value equal to the large early release frequency (LERF) criterion (1×10^{-5} per year) was recommended for the PPG. By maintaining the frequency of events leading to uncovery of the spent fuel at decommissioning facilities below the PPG, the risk from zirconium fires will be low and consistent with the guidance in RG 1.174 for allowing changes to the plant licensing basis that slightly increase risk. With one exception (the H.B. Robinson site) all Central and Eastern sites which implement the IDCs and SDAs would be expected to meet the PPG regardless of whether LLNL or EPRI seismic hazard estimates are assumed. The Robinson site would satisfy the PPG if the EPRI hazard estimate is applied but not if the LLNL hazard is used. Therefore, Western sites and Robinson would need to be considered on a site-specific basis because of important differences in seismically induced failure potential of the SFPs.

The appropriateness of the PPG was questioned by the ACRS in view of potential effects of the fission product ruthenium, the release of fuel fines, and the effects of revised plume parameters. The staff added sensitivity studies to its analyses to examine these issues. The consequences of a significant release of ruthenium and fuel fines were found to be notable, but not so important as to render inappropriate the staff's proposed PPG of $1x10^{-5}$ per year. The plume parameter sensitivities were found to be of lesser significance.

In its thermal-hydraulic analysis the staff concluded that it was not feasible, without numerous constraints, to establish a generic decay heat level (and therefore a decay time) beyond which a zirconium fire is physically impossible. Heat removal is very sensitive to these additional constraints, which involve factors such as fuel assembly geometry and SFP rack configuration. However, fuel assembly geometry and rack configuration are plant specific, and both are subject to unpredictable changes after an earthquake or cask drop that drains the pool. Therefore, since a non-negligible decay heat source lasts many years and since configurations ensuring sufficient air flow for cooling cannot be assured, the possibility of reaching the zirconium ignition temperature cannot be precluded on a generic basis.

The staff found that the event sequences important to risk at decommissioning plants are limited to large earthquakes and cask drop events. For emergency planning (EP) assessments this is an important difference relative to operating plants where typically a large number of different sequences make significant contributions to risk. Relaxation of offsite EP a few months after shutdown resulted in only a "small change" in risk, consistent with the guidance of RG 1.174. The change in risk due to relaxation of offsite EP is small because the overall risk is low, and because even under current EP requirements, EP was judged to have marginal impact on evacuation effectiveness in the severe earthquakes that dominate SFP risk. All other sequences including cask drops (for which emergency planning is expected to be more effective) are too low in likelihood to have a significant impact on risk. For comparison, at operating reactors additional risk-significant accidents for which EP is expected to provide dose savings are on the order of $1x10^{-5}$ per year, while for decommissioning facilities, the largest contributor for which EP would provide dose savings is about two orders of magnitude lower (cask drop sequence at $2x10^{-7}$ per year). Other policy considerations beyond the scope of this technical study will need to be considered for EP requirement revisions and previous exemptions because a criteria of sufficient cooling to preclude a fire cannot be satisfied on a generic basis.

Insurance does not lend itself to a "small change in risk" analysis because insurance affects neither the probability nor the consequences of an event. The study found that as long as a zirconium fire is possible, the long-term consequences of an SFP fire may be significant. These long-term consequences (and risk) decrease very slowly because cesium-137 has a half life of approximately 30 years. The thermal-hydraulic analysis indicates that when air flow has been restricted, such as might occur after a cask drop or major earthquake, the possibility of a fire lasts many years and a criterion of "sufficient cooling to preclude a fire" can not be defined on a generic basis. Other policy considerations beyond the scope of this technical study will therefore need to be considered for insurance requirements.

The study also discusses implications for security provisions at decommissioning plants. For security, risk insights can be used to determine what targets are important to protect against sabotage. However, any revisions in security provisions should be constrained by an effectiveness assessment of the safeguards provisions against a design-basis threat. Because

the possibility of a zirconium fire leading to a large fission product release cannot be ruled out even many years after final shutdown, the safeguards provisions at decommissioning plants should undergo further review. The results of this study may have implications on previous exemptions at decommissioning sites, devitalization of spent fuel pools at operating reactors and related regulatory activities.

The staff's risk analyses were complicated by a lack of data on severe-earthquake return frequencies, source term generation in an air environment, and SFP design variability. Although the staff believes that decommissioning rulemaking can proceed on the basis of the current assessment, more research may be useful to reduce uncertainties and to provide insights on operating reactor safety. In particular, the staff believes that research may be useful on source term generation in air, which could also be important to the risk of accidents at operating reactors during shutdowns, when the reactor coolant system and the primary containment may both be open.

In summary, the study finds that:

1. The risk at decommissioning plants is low and well within the Commission's safety goals. The risk is low because of the very low likelihood of a zirconium fire even though the consequences from a zirconium fire could be serious.

2. The overall low risk in conjunction with important differences in dominant sequences relative to operating reactors, results in a small change in risk at decommissioning plants if offsite emergency planning is relaxed. The change is consistent with staff guidelines for small increases in risk.

3. Insurance, security, and EP requirement revisions need to be considered in light of other policy considerations because a criterion of "sufficient cooling to preclude a fire" cannot be satisfied on a generic basis.

4. Research on source term generation in an air environment would be useful for reducing uncertainties.

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7.0 ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ANSI	American National Standard Institute
ANS	American Nuclear Society
ASB	NRC Auxiliary Systems Branch (Plant Systems Branch)
atm	atmosphere
BNL	Brookhaven National Laboratory
BTP	branch technical position
BWR	boiling-water reactor
CFD	computational fluid dynamics
CFM	cubic feet per minute
CFR	*U.S. Code of Federal Regulations*
DOE	Department of Energy
DSP	decommissioning status plant
ECCS	emergency core cooling system
EOF	emergency operations facility
EP	emergency plan
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
ET	event tree
FFU	frequency of fuel uncovery
FT	fault tree
gpm	gallon(s) per minute
GI	generic issue
GWD	gigawatt-day
HCLPF	high confidence in low probability of failure
HRA	human reliability analysis
HVAC	heating, ventilation, and air conditioning
IDC	industry decommissioning commitment
INEEL	Idaho National Engineering and Environmental Laboratory
ISFSI	independent spent fuel pool installation
kW	kilowatt
LERF	large early release frequency
LLNL	Lawrence Livermore National Laboratory
LOSP	loss of offsite power
LWR	light-water reactor

MR	Maintenance Rule
MW	megawatt
MWD	megawatt-day
MTU	metric ton uranium
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	NRC Office of Nuclear Reactor Regulation
OSC	operations support center
POE	probability of exceedance
POF	probability of failure
PPG	pool performance guideline
PRA	probabilistic risk assessment
PWR	pressurized-water reactor
QA	quality assurance
QHO	quantitative health objective
RES	NRC Office of Regulatory Research
RG	regulatory guide
SDA	staff decommissioning assumption
SF	spent fuel
SFP	spent fuel pool
SFPC	spent fuel pool cooling system
SFPCC	spent fuel pool cooling and cleaning system
SHARP	Spent Fuel Heatup Analytical Response Program
SNL	Sandia National Laboratory
SRM	staff requirements memorandum
SRP	standard review plan
SSC	systems, structures, and components
SSE	safe shutdown earthquake
TS	technical specification
TSC	technical support center
UKAEA	United Kingdom Atomic Energy Authority
WIPP	waste isolation pilot plant

NUREG/CR-4982
BNL-NUREG-52093

Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82

Manuscript Completed: June 1987
Date Published: July 1987

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ABSTRACT

This investigation provides an assessment of the likelihood and consequences of a severe accident in a spent fuel storage pool - the complete draining of the pool. Potential mechanisms and conditions for failure of the spent fuel, and the subsequent release of the fission products, are identified. Two older PWR and BWR spent fuel storage pool designs are considered based on a preliminary screening study which tried to identify vulnerabilities. Internal and external events and accidents are assessed. Conditions which could lead to failure of the spent fuel Zircaloy cladding as a result of cladding rupture or as a result of a self-sustaining oxidation reaction are presented. Propagation of a cladding fire to older stored fuel assemblies is evaluated. Spent fuel pool fission product inventory is estimated and the releases and consequences for the various cladding failure scenarios are provided. Possible preventive or mitigative measures are qualitatively evaluated. The uncertainties in the risk estimate are large, and areas where additional evaluations are needed to reduce uncertainty are identified.

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PREFACE

This study is an initial attempt by Brookhaven National Laboratory to characterize the radiological risks posed by storage of spent reactor fuel at commercial reactor sites in the United States. This work was done at the request of the U.S. Nuclear Regulatory Commission in support of their technical analysis related to Generic Safety Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools." The method of analysis used in this study was to a) survey the spent fuel pool configurations at commercial reactor sites in terms of the characteristics that are important to risk and b) perform detailed analyses of those spent fuel configurations for which the risk appeared to be potentially significant. The detailed analyses were performed by using the methodology of probabilistic risk assessment that has been used extensively in the assessment of power plant risks during normal operation. Thus, this initial study, while limited in resources, required the integration of several technologically distinct disciplines (e.g., seismic analysis, fuel degradation analysis, offsite consequence analysis). Although these disciplines have been integrated before in the normal operation risk assessments, the application to the spent fuel problem posed novel and uncertain conditions not encountered in the normal operation risk assessments. The present study did not address: the potential for recriticality; the fuel damage process during a slow pool drainage; and the fuel reconfiguration after a clad fire. The results of this study have additional uncertainty, beyond those characteristic of traditional risk assessment studies for reactor operations, which is associated with the novel aspects of the phenomenology and the limitations of the data base.

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ACKNOWLEDGEMENTS

This work was performed for the Reactor and Plant Safety Issues Branch of the Division of Reactor and Plant Systems, NRC/RES. The NRC Managers for the program were Mr. E. Throm and Dr. M. Wohl who provided considerable input and technical direction to the program. Mr. E. Throm also assisted by coordinating a thorough NRC review of the initial draft of this report.

As with most integrated programs technical contributions were provided by many people within and external to BNL. In particular, the authors are indebted to Drs. A. Benjamin (SNL) and F. Best (Texas A&M) who provided considerable assistance in implementing and understanding the SFUEL code. The authors are also grateful for several technical contributions from the DNE staff at BNL. Dr. K. Shiu provided considerable assistance in evaluating the seismic hazard. Dr. T. Teichman assisted in several statistical evaluations. Dr. M. Reich and Dr. J. Pires were was especially helpful in the interpretation of pool structural fragility results and Dr. L. Teutonico provided an evaluation of the oxidation rate data. Dr. A. Tingle helped set up and interpret the consequence calculations with the CRAC2 code. Mr. A. Aronson implemented the ORIGEN2 code and provided the calculations for spent fuel pool fission product inventories for the actual discharge histories. Drs. W. Pratt and R. Bari provided administrative assistance and were very helpful in providing a thorough technical review of the final report.

The authors are especially grateful to Ms. S. Flippen for her excellent typing of this report and for cheerfully accepting the numerous additions and revisions to this manuscript.

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EXECUTIVE SUMMARY

S.1 INTRODUCTION

Generic Safety Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools," was assigned a MEDIUM priority in November 1983.[1] In this prioritization, the NRC staff considered three factors that had not been included in earlier risk assessments:[2]

1. Spent fuel is currently being stored rather than shipped for reprocessing or repository disposal, resulting in much larger inventories of spent assemblies in reactor fuel basins than had previously been anticipated,

2. In order to accommodate the larger inventory, high density racking is necessary, and

3. A theoretical model[3,4] suggested the possibility of Zircaloy fire, propagating from recently discharged assemblies to lower power assemblies in the event of complete drainage of water from the pool.

The Reactor Safety Study,[2] commonly referred to as WASH-1400, concluded that the risks associated with spent fuel storage were extremely small in comparison with accidents associated with the reactor core. That conclusion was based on design and operational features of the storage pools which made the loss of water inventory highly unlikely. In addition it was assumed that the pool inventory would be limited to about one-third of a core.

Subsequent to the Reactor Safety Study, A.S. Benjamin et al.[3,4] investigated the heatup of spent fuel following drainage of the pool. A computer code, SFUEL, was developed to analyze thermal-hydraulic phenomena occurring when storage racks and spent assemblies become exposed to air.

Calculations with SFUEL indicated that, for some storage configurations and decay times, the Zircaloy cladding could reach temperatures at which the exothermic oxidation would become self-sustaining with resultant destruction of the cladding and fission product release. The possibility of propagation to adjacent assemblies (i.e., the cladding would catch fire and burn at a high enough temperature to heat neighboring fuel assemblies to the ignition point) was also identified. Under certain conditions, the entire inventory of stored fuel could become involved. Cladding fires of this type could occur at temperatures well below the melting point of the UO_2 fuel. The cladding ignition point is about 900°C compared to the fuel melting point of 2880°C.

There is no case on record of a significant loss of water inventory from a domestic, commercial spent fuel storage pool. However, two recent incidents have raised concern about the possibility of a partial draindown of a storage pool as a result of pneumatic seal failures.

The first incident occurred at the Haddam Neck reactor during preparations for refueling with the refueling cavity flooded. An inflatable seal bridging the annulus between the reactor vessel flange and the reactor cavity bearing plate extruded into the gap, allowing 200,000 gallons of borated water to drain out of the refueling cavity into the lower levels of the containment

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building in about 20 minutes. Gates to the transfer tube and the fuel storage pool were in the closed position, so no water drained from the pool.[6]

The second pneumatic seal failure incident occurred in the Hatch spent storage pool/transfer canal, (the seal failure at Hatch was not in the refueling cavity) which released approximately 141, 000 gallons of water and resulted in a drop in water level in the pool of about five feet.[7]

However, the BNL review of these events indicates that they are unique to the plants involved and such events are unlikely to cause a substantial loss of pool inventory for other plants. However, pneumatic seal failures may expose individual fuel bundles during refueling and these events are being investigated as part of Generic Issue 137, "Refueling Cavity Seal Failure."

S.2 OBJECTIVE

The objective of this investigation is to provide an assessment of the potential risk from possible accidents in spent fuel pools. The risks are defined in terms of:

- the probabilities of various initiating events that might compromise the structural integrity of the pool or its cooling capability,
- the probability of a system failure, given an initiating event,
- fuel failure mechanisms, given a system failure,
- potential radionuclide releases, and
- consequences of a specified release.

This study generally follows the logic of a typical probabilistic risk analysis (PRA); however, because of the relatively limited number of potential accident sequences which could result in the draining of the pool, the analyses have been greatly simplified.

The configurations of spent fuel storage pools vary from plant to plant. In BWR's, the pools are located within the reactor building with the bottom of the pool at about the same elevation as the upper portion of the reactor pressure vessel. During refueling the cavity above the top of the pressure vessel is flooded to the same elevation as the storage pool, so that fuel assemblies can be transferred directly from the reactor to the pool via a gate which separates the pool from the cavity. In PWR plants, the storage pool is located in an auxiliary building. In some cases the pool surface is at about grade level, in others the pool bottom is at grade. The refueling cavities are usually connected to the storage pool by a transfer tube. During refueling the spent assembly is removed from the reactor vessel and placed in a container which then turns on its side, moves through the transfer tube to the storage pool, is set upright again and removed from the transfer container to a storage rack. Various gates and weirs separate different sections of the transfer and storage systems. A screening study was performed to identify potentially risk significant sequences involving spent fuel pools, the pool design features of the commercial power plants were reviewed and summarized.

In order to prioritize the present risk analysis, a preliminary risk assessment was performed for spent fuel pools using the RSS methodology[2] and the results of the above screening study. This preliminary study indicated that a seismic initiated failure of the pool was the dominant risk

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contributor. Based on this assessment, two older BWR and PWR plants were selected for more detailed studies based on their perceived vulnerability to seismic events. Specifically, Millstone 1 and Ginna, were selected because of availability of data, fuel pool inventory, and the relative familiarity of the BNL staff with the various candidate sites. The operating histories of the two plants were modeled to obtain a realistic radioactive inventory in the various spent fuel batches.

S.3 ACCIDENT INITIATING EVENTS AND PROBABILITY ESTIMATES

Accident initiating events that have been considered include both internal and external events:

- pool heatup due to loss of cooling water circulation capability,
- structural failure of pool due to seismic events or missiles,
- partial draindown of pool due to pneumatic seal failure, and
- structural failure of pool due to a heavy load drop.

Accidents leading to complete pool draining that might be initiated by loss of cooling water circulation capability, missiles, and pneumatic seal failure were found to have a very low likelihood. However, the frequency estimates for pool draining due to structural failure resulting from seismic events and heavy load drops were found to be quite uncertain. In the case of seismic events, the seismic hazard and structural fragilities both contribute to the uncertainty range. For heavy load drops, human error probabilities, structural damage potentials and recovery actions are the primary sources of uncertainties.

S.4 EVALUATION OF FUEL CLADDING FAILURE

The SFUEL computer code developed at Sandia National Laboratories (SNL) by Benjamin et al.,[3] analyzes the behavior of spent fuel assemblies after an accident has drained the pool. The analyses predict that self-sustaining oxidation of the Zircaloy cladding (i.e., a cladding fire) would occur for a wide range of decay heat levels and storage geometries. Several limitations in the SFUEL analyses had been recognized in Reference 3 and have been addressed in a modified version of the code, SFUEL1W.[4]

The BNL evaluations of SFUEL1W have led to the conclusions that the modified code gives a reasonable estimate of the potential for propagation of a cladding fire from high power to low power spent fuel while the fuel is intact. The code therefore provides a valuable tool for assessing the likelihood of a cladding fire for a variety of intact spent fuel configurations in the event that the pool is drained.

S.5 CONSEQUENCE EVALUATION

Radioactive releases are estimated for the two plants for cladding failure scenarios predicted by SFUEL calculations involving cladding fires. Partial drainage events where the cladding may rupture and release the rod gap inventory (without a fire) are also presented.

The radioactive inventories contained in the spent fuel pools (as of April 1987) for Millstone 1 and Ginna were calculated using the ORIGEN2

computer code,[8] based on the operating histories of each of the plants (Appendix A). The calculated data included the 1987 inventories for each fuel batch discharged at each refueling over the operating history.

Fractional releases for various groups of radionuclides were estimated based on the physical parameters characterizing the SFUEL failure scenario. Thus, source terms were estimated corresponding to the seven accident scenarios: five involving cladding fire for various amounts of fuel, and two involving cladding rupture (without a fire).

Off-site radiological consequences were calculated using the CRAC2 computer code.[9] Because of several features in the health physics modeling in the CRAC2 code, the population dose results are not very sensitive to the estimated fission product release. A more sensitive measure of the accident severity appears to be the interdiction area (contaminated land area) which in the worst cases was about two hundred square miles. While the long-term health effects (i.e., person-rem) are potentially large, it is important to note that no "prompt fatalities" were predicted and the risk of injury was also negligible.

S.6 RISK PROFILE

The likelihood and consequences of various spent fuel pool accidents have been combined to obtain the risks which are summarized in Table S.1. The population dose results are insensitive to the fission product release because they are driven by decontamination levels assigned within the CRAC2 code. The health physics models in CRAC2 assign a maximum allowable dose for each individual before the contaminated area is reoccupied. This allowable dose for the returning population is the dominant contributor to total exposure and limits the utility of the dose calculation. Thus the land interdiction area is included in Table S.1 as a more sensitive representation of the severity of the postulated accident.

The unique character of fuel pool accidents (potentially large releases of long lived isotopes) makes it difficult to compare directly to reactor core melt accidents. There are no early health effects. The long-term exposure calculations are driven by assumptions in the CRAC modeling and the results are not very sensitive to the severity of the accident. There is substantial uncertainty in the fission product release estimates. These uncertainties are due to both uncertainty in the accident progression (fuel temperature after clad oxidation and fuel relocation occurs) and the uncertainty in fission product decontamination.

S.7 CONSIDERATION OF MEASURES WHICH MIGHT REDUCE CONSEQUENCES

A number of potential preventive and mitigative measures were identified, but because of the large uncertainty ranges in Table S.1, the potential benefits of such measures are also uncertain and plant specific. A cost benefit analysis has not been performed. Rather, the phenomenological insights, developed during the investigation, have been used to generate a list of possible risk reduction measures. Calculations with the SFUEL code indicate that, for those plants that use a high density storage rack configuration, a factor of five reduction in the fire probability (given loss of pool inventory) can be achieved by improved air circulation capability. This reduction

factor is based upon the time period after discharge for which SFUEL predicted that the decay heat is sufficient to initiate a clad fire. Considering the large uncertainty in risk, a plant specific cost/benefit analyses should be performed before such risk reduction measures are implemented.

S.8 CONCLUSIONS

This limited risk assessment, which was performed for two older spent fuel pools, indicates that the risk estimates are quite uncertain and could potentially (under worst case assumptions) be significant. The uncertainty in risk is dominated by the estimated uncertainty in the likelihood of the loss of pool integrity due to beyond design basis seismic events. This uncertainty is, in turn, driven by the uncertainty in the seismic hazard and the spent fuel pool fragility. These risk ranges are consistent with the current medium priority assigned to the issue by the NRC.[1] It is not clear that these uncertainty ranges are directly applicable to other plants because the plants selected for detailed study were chosen specifically for their perceived vulnerability to seismic events after an extensive screening process (refer to Section S.2). For example, if the fragility estimates for plants, which meet the new seismic design criteria, were used, a significant reduction in the predicted likelihood of seismically initiated pool failure would result. In addition many of the new plants have pool configurations and administrative procedures which would preclude cask drop accidents. Therefore, in order to determine whether other plants have a significant risk profile, a plant specific evaluation would be required. A key part of such an evaluation would be to obtain a realistic seismic fragility estimate for the specific spent fuel pool.

S.9 REFERENCES FOR SUMMARY

1. "A Prioritization of Generic Safety Issues," Division of Safety Technology, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, NUREG-0933, December 1983, pp. 3.82-1 through 6.

2. "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, NUREG-75/014 (WASH-1400), October 1975, App. I, Section 5.

3. A.S. Benjamin, D.J. McClosksy, D.A. Powers, and S.A. Dupree, "Spent Fuel Heatup Following Loss of Water During Storage," prepared for the U.S. Nuclear Regulatory Commission by Sandia Laboratories, NUREG/CR-0649 (SAND77-1371), May 1979.

4. N.A. Pisano, F. Best, A.S. Benjamin and K.T. Stalker, "The Potential for Propagation of a Self-Sustaining Zirconium Oxidation Following Loss of Water in a Spent Fuel Storage Pool," prepared for the U.S. Nuclear Regulatory Commission by Sandia Laboratories, (Draft Manuscript, January 1984) (Note: the project ran out of funds before the report was published.)

5. IE Bulletin No. 84-03: "Refueling Cavity Water Seal," U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, August 24, 1984.

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6. Licensee Event Report, LER No. 84-013-00, Haddam Neck, Docket No. 50-213, "Failure of Refueling Pool Seal," 09/21/84.

7. Nucleonics Week, December 11, 1986, pg. 3-4.

8. A.G. Croff, "ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Composition and Characteristics of Nuclear Materials," Nuclear Technology, Vol. 62, pp. 335-352, September 1983.

9. L.T. Ritchie, J.D. Johnson and R.M. Blond, Calculations of Reactor Accident Consequences Version 2, CRAC2: Computer Code User's Guide, prepared by Sandia National Laboratories for the U.S. Nuclear Regulatory Commission, NUREG/CR-2326 (SAND81-1994), February 1983.

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Table S.1 Estimated Risk for the Two Spent Fuel Pools from the Two Dominant Contributors

Accident Initiator	Spent Fuel Pool Fire Probability/Ry	Health Risk[1] (Man-rem/Ry)	Interdiction[1] Risk (Sq. Mi./Ry)
Seismic induced PWR pool failure	2.6×10^{-4}-1.6×10^{-10}	600-Neg.*	.011-Neg.
Seismic induced BWR pool failure	6.5×10^{-5}-4×10^{-11}	156-Neg.	.003-Neg.
Cask drop[2] induced PWR pool failure	3×10^{-5}-3×10^{-12}	70-Neg.	.001-Neg.
Cask drop[2] induced BWR pool failure	8×10^{-6}-8×10^{-13}	20-Neg.	4×10^{-4}-Neg.

*Neg. - Negligible.

[1]The upper end of the risk ranges assumes no fire propagation from the last fuel discharge to older fuel. However, the fission products in the last fuel discharge were assumed to be released during the fire with no fission product decontamination on structures.

[2]After removal of accumulated inventory resumes. Presently, most plants are accumulating spent fuel in the pool without shipping to permanent storage. (Note that many new plants have pool configurations and administrative procedures which would preclude this failure mode.)

1. INTRODUCTION

Generic Safety Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools," was assigned a MEDIUM priority in November 1983.[1] In this prioritization, the NRC staff considered three factors that had not been included in earlier risk assessments:[2]

1. Spent fuel is currently being stored rather than shipped for reprocessing or repository disposal, resulting in much larger inventories of spent assemblies in reactor fuel basins than had previously been anticipated;

2. In order to accommodate the larger inventory, high density racking is necessary, and

3. A theoretical model[3] suggested the possibility of Zircaloy fire, propagating from assembly to assembly in the event of complete drainage of water from the pool.

1.1 Previous Investigations

The Reactor Safety Study,[2] commonly referred to as WASH-1400, concluded that the risks associated with spent fuel storage were extremely small in comparison with accidents associated with the reactor core. That conclusion was based on design and operational features of the storage pools which made the loss of water inventory highly unlikely, e.g.,

- The pool structures were designed to withstand safe shutdown earthquakes,
- The fuel racks were designed to preclude criticality,
- Pool design and instrumentation precluded inadvertent and undetected loss of water inventory,
- Procedures and interlocks prevented the drop of heavy loads on stored assemblies, and
- The storage structures were designed to accommodate the forces and missiles generated by violent storms.

Probabilities of pool failures due to external events (earthquakes, missiles) or heavy load drops were estimated to be in the range of 10^{-6}/year. Radioactive release estimates were based on melting of 1/3 of a core for various decay periods, with and without filtration of the building atmosphere (see Ref. 2, Table I 5-2).

Subsequent to the Reactor Safety Study, A.S. Benjamin et al.[3] investigated the heatup of spent fuel following drainage of the pool. A computer code, SFUEL, was developed to analyze thermal-hydraulic phenomena occurring when storage racks and spent assemblies become exposed to air. The computer model takes into account decay time, fuel assembly design, storage racks design, packing density, room ventilation and other variables that affect the heatup of the fuel.

Calculations with SFUEL indicated that, for some storage configurations and decay times, the Zircaloy cladding could reach temperatures at which the exothermic oxidation would become self-sustaining with resultant destruction

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of the cladding and fission product release. The possibility of propagation to adjacent assemblies (i.e., the cladding would catch fire and burn at a hot enough temperature to heat neighboring fuel assemblies to the ignition point) was also identified. Under certain conditions, the entire inventory of stored fuel could become involved. Cladding fires of this type could occur at temperatures well below the melting point of the UO_2 fuel. The cladding ignition point is about 900°C compared to the fuel melting point of 2880°C.

Uncertainties in the SFUEL calculations were primarily attributed to uncertainties in the zirconium oxidation rates.

Further work was done to refine the SFUEL computer model and to compare calculated results with experimental data.[4] These more recent results have generally confirmed the earlier concepts of a Zircaloy fire which, given the right conditions, will propagate to neighboring assemblies. However, comparisons to out-of-pile heat-up data have not shown good agreement with the code. As discussed in Section 3, the SNL authors noted that more work in several areas was needed to define more precisely the conditions and configurations which allow or prevent propagation.

Several studies have been conducted on alternative spent fuel storage concepts. Among these is a report published by the Electric Power Research Institute (EPRI), which applies probabilistic risk assessment techniques to several storage concepts.[5] While this study does not directly address Generic Safety Issue 82, it does provide useful insight on appropriate analytical methodology as well as useful data on an in-ground (on-site) storage pool.

1.2 Related Events

There is no case on record of a significant loss of water inventory from a domestic, commercial spent fuel storage pool. However, two recent incidents have raised concern about the possibility of a partial draindown of a storage pool as a result of pneumatic seal failures.

The first incident occurred at the Haddam Neck reactor during preparations for refueling.[6] An inflatable seal bridging the annulus between the reactor vessel flange and the reactor cavity bearing plate extruded into the gap, allowing 200,000 gallons of borated water to drain out of the refueling cavity into the lower levels of the containment building in about 20 minutes. Gates to the transfer tube and the fuel storage pool were in the closed position, so no water drained from the pool.[7] Had these gates been open at the time of the leak, and had they not been closed within 10 to 15 minutes, the pool would have drained to a depth of about 8.5 feet, exposing the upper 3 feet of the active fuel region in the spent fuel assemblies.[7] Also, had the transfer of spent fuel been in progress with an assembly on the refueling machine, immediate action would have been necessary to place the assembly in a safe location under water to limit exposure to personnel. The NRC has identified this aspect of a seal failure accident as potential Generic Issue 137, "Refueling Cavity Seal Failure."[8] The current schedule for evaluation of the issue is December 1987.

The NRC Office of Inspection and Enforcement required all licensees to promptly evaluate the potential for refueling cavity seal failures.[6] Responses indicated that the refueling cavity configuration at Haddam Neck is

unique in that the annulus between the reactor flange and the cavity bearing plate is more than 2 feet wide. In most plants this gap is only about 2 inches wide.[9] About 40 operating (or soon to operate) reactors use inflatable seals in the refueling cavity. However, because of design differences, the Haddam Neck failure does not appear to be directly applicable to the other plants. It is noted that most BWR plants have permanent steel bellows seals to fill the gap between the reactor flange and the cavity bearing plate. This issue is discussed more fully in Section 2.3.

The second pneumatic seal failure incident occurred in the Hatch spent storage pool/transfer canal in December 1986.[10] In this incident, a pair of pneumatic seals deflated when the compressed air supply was inadvertently shut off. The seals involved were in the transfer canal flexible seismic joint. The leak detection annunciator failed to alarm and the leak was not discovered for about 7-1/2 hours. Approximately 141,000 gallons of water leaked from the storage fuel and the water level dropped about 5-1/2 feet.

1.3 Risk Potential

This study addresses beyond design basis accidents in spent fuel pools that might result in the complete loss of pool water due to structural failure, massive leaks or boil-off of inventory due to prolonged failure of cooling systems. The risk potentials are defined in terms of

- the probabilities of various initiating events that might compromise the structural integrity of the pool or its cooling capability,
- the probability of a system failure, given an initiating event,
- fuel failure mechanisms, given a system failure,
- potential radionuclide releases, and
- consequences of a specified release.

The analyses generally follow the logic of typical probabilistic risk analyses (PRA); however, because of the relatively limited number of potential accident sequences, which could result in the draining of the pool, the analyses have been greatly simplified.

1.4 Discussion of Spent Fuel Storage Pool Designs and Features

The general design criteria for spent fuel storage facilities are stated in Appendix A of 10 CFR 50,[11] and are discussed more fully in Regulatory Guide 1.13.[12]

The pool structures, spent fuel racks and overhead cranes must be designed to Seismic Category I standards. It is required that the systems be designed (1) with capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.[11]

As part of the preliminary screening study for accident vulnerabilities, the design features of the spent fuel pools for the commercial power plants

were reviewed and assembled. The configurations of spent fuel storage pools vary from plant to plant. Table 1.1 summarizes this information for each of the pools.

In BWRs the pools are located within the reactor building with the bottom of the pool at about the same elevation as the upper portion of the reactor pressure vessel. (For example, at Oyster Creek the bottom of the pool is at elevation 80'6", and the top at 119'3". The water depth is 38 feet.) During refueling, the cavity above the top of the pressure vessel is flooded to the same elevation as the storage pool, so that fuel assemblies can be transferred directly from the reactor to the pool via a gate which separates the pool from the cavity.

In PWR plants the storage pool is located in an auxiliary building. In some cases the pool surface is at about grade level, in others the pool bottom is at grade. The refueling cavities are usually connected to the storage pool by a transfer tube. During refueling the spent assembly is removed from the reactor vessel and placed in a container which then turns on its side, moves through transfer tube to storage pool, set upright again and removed from the transfer container to a storage rack. Various gates and weirs separate different sections of the transfer and storage systems. More details concerning various configurations are given in Section 2.3.

1.5 More Detailed Studies

The overall objective of the present investigation was to determine whether possible severe accidents involving spent fuel pools posed a significant risk to the public. In order to prioritize the investigation a preliminary risk assessment was performed using RSS[2] methodology to identify the potentially important accident sequences and the characteristics of specific fuel pools which could lead to unusually high vulnerability to accidents. This preliminary risk assessment indicated that seismically induced structural failure of the pool appeared to dominate the spent fuel pool risk. This appeared to be particularly true for older plants in the eastern states where recent studies have indicated an increase in the estimated seismic hazard. Based on this preliminary study, two older BWR and PWR plants were selected for more detailed studies because of their perceived vulnerability to seismic events. Specifically, Millstone 1 and Ginna, were selected because of availability of data, fuel pool inventory, and the relative familiarity of the BNL staff with the various candidate sites. The operating histories of the two plants were modeled to obtain a realistic radioactive inventory in the various spent fuel batches. Details of the modeling procedures and a listing of the calculated radionuclide content are presented in Appendix A.

It should be noted that both plants have relatively large inventories of spent fuel assemblies in their spent fuel pools.

1.6 Report Content

Accident initiating events and their probabilities are covered in Section 2. Fuel cladding failure scenarios based on the SFUEL1W Computer Code are evaluated in Section 3. Included are sensitivity analyses of the failure scenarios arising from uncertainties in Zircaloy oxidation reaction rate data, and hardware configuration assumptions. Section 4 presents data on the

potential for releases of radionuclides under various cladding failure scenarios and compares the projected releases with releases associated with severe core accident sequences. In Section 5, risk profiles are developed in terms of person-rem population doses for several accident sequences. Section 6 considers measures that might mitigate pool draining and/or Zircaloy fire propagation.

1.7 References for Section 1

1. "A Prioritization of Generic Safety Issues," Division of Safety Technology, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, NUREG-0933, December 1983, pp. 3.82-1 through 6.

2. "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, NUREG-75/014 (WASH-1400), October 1975, App. I, Section 5.

3. A.S. Benjamin, D.J. McClosksy, D.A. Powers, and S.A. Dupree, "Spent Fuel Heatup Following Loss of Water During Storage," prepared for the U.S. Nuclear Regulatory Commission by Sandia Laboratories, NUREG/CR-0649 (SAND77-1371), May 1979.

4. N.A. Pisano, F. Best, A.S. Benjamin and K.T. Stalker, "The Potential for Propagation of a Self-Sustaining Zirconium Oxidation Following Loss of Water in a Spent Fuel Storage Pool," prepared for the U.S. Nuclear Regulatory Commission by Sandia Laboratories, (Draft Manuscript, January 1984) (Note: the project ran out of funds before the report was published.)

5. D.D. Orvis, C. Johnson, and R. Jones, "Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment," prepared for the Electric Power Research Institute by the NUS Corporation, EPRI NP-3365, February 1984.

6. IE Bulletin No. 84-03: "Refueling Cavity Water Seal," U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, August 24, 1984.

7. Licensee Event Report, LER No. 84-013-00, Haddam Neck, Docket No. 50-213, "Failure of Refueling Pool Seal," 09/21/84.

8. "Generic Issue Management Control System - First Quarter FY-87 Updates," Memorandum from T.P. Speis, Director, Division of Safety Review and Oversight, to H.R. Denton, Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, February 13, 1987.

9. Licensee Responses to NRC IE Bulletin No. 84-03.

10. U.S. Nuclear Regulatory Commission, "Morning Report - Region II," December 5, 1986.

11. Code of Federal Regulations, Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities, Appendix A, 'General Design Criteria for Nuclear Power Plants,' General Design Criterion 61, 'Fuel Storage and Handling and Radioactivity Control'."

12. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," December 1981.

Table 1.1 BWR's: DATA ON SPENT FUEL STORAGE BASINS. Included are spent fuel storage inventories as of December 1984; fractions of core in storage, comparisons with the "reference case" of radionuclide inventory, locations of spent fuel basins, and seismic design bases of pools.

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[a]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Big Rock Point	240	84	172	2.05	4.9	AB, grd	DBE=0.05g
Browns Ferry-1	3293	764	1068	1.40	46.1	RB, ele	DBE=0.20g
Browns Ferry-2	3293	764	889	1.16	38.2	RB, ele	DBE=0.20g
Browns Ferry-3	3293	764	1768	2.31	76.1	RB, ele	DBE=0.20g
Brunswick-1	2436	560	1056[f]	1.89	46.0	RB, ele	DBE=0.16g
Brunswick-2	2436	560	924[g]	1.65	40.2	RB, ele	DBE=0.16g
Cooper	2381	548	985	1.80	42.9	RB, ele	DBE=0.2g
Dresden-1	700	464	221	0.48	3.36		DBE=0.20g
Dresden-2	2527	724	2014[h]	2.78[h]	70.3[h]	RB, ele	DBE=0.2g
Dresden-3	2527	724	-	-	-	RB, ele	DBE=0.2g
Duane Arnold	1658	368	576	1.57	26.0	RB, ele	DBE=0.12g
Fitzpatrick	2436	560	816	1.46	35.6	RB, ele	DBE=0.15g
Grand Gulf-1	3833	N/A[i]	0	0.00	0.0	N/A	
Hatch-1	2436	560	140	0.25	6.1	RB, ele	DBE=0.15g

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Table 1.1 (Cont'd)

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[a]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Hatch-2	2436	560	1284	2.29	55.8	RB, ele	DBE=0.15g
Humboldt Bay	220	172	251	1.46	3.2	N/A	DBE=0.50g
LaCrosse	165	72	207	2.88	4.8	AB, grd	DBE=0.12g
LaSalle-1	3323	N/A	0	0.00	0.0	RB, ele	SSE=0.20g
LaSalle-2	3323	N/A	0	0.00	0.0	RB, ele	SSE=0.20g
Limerick-1	3293	N/A	0	0.00	0.0	RB, ele	SSE=0.13g
Millstone-1	2011	580	1346	2.32	46.7	RB, ele	DBE=0.17g
Monticello	1670	484	1137	2.35	39.2	RB, ele	DBE=0.12g
Nine Mile Point-1	1850	532	1244	2.34	43.3	RB, ele	DBE=0.11g
Oyster Creek	1930	560	1375	2.46	47.5	RB, ele	DBE=0.22g
Peach Bottom-2	3293	764	1361	1.78	58.6	RB, ele	DBE=0.12g
Peach Bottom-3	3293	764	1212	1.59	52.4	RB, ele	DBE=0.12g
Pilgrim-1	1998	580	1128	1.94	38.8	RB, ele	DBE=0.15g
Quad Cities-1	2511	724	1730	2.39	60.0	RB, ele	DBE=0.24g
Quad Cities-2	2511	724	412	0.57	14.3	RB, ele	DBE=0.24g

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Table 1.1 (Cont'd)

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[a]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Susquehanna-1	3293	764	0	0.00	0.0	RB, ele	SSE=0.1g
Susquehanna-2	3293	764	0	0.00	0.0	RB, ele	SSE=0.1g
Vermont Yankee	1593	368	1174	3.19	50.8	RB, ele	DBE=0.14g
Wash. Nucl.-2	3323	N/A	0	0.00	0.0	N/A	SSE=0.32g

Footnotes

a) Source: U. S. Nuclear Regulatory Commission, Licensed Operating Reactors, NUREG-0020, Vol. 9, No. 1, January 1985.

b) (Stored Assemblies)/(Assemblies in Core).

c) "Reference Source Term" assumes a thermal power of 3000 MWt, stored inventory from ten annual discharges, last discharge six months ago, total inventory 1750 assemblies. Source term relative to "Reference Source Term" has not been corrected for age of fuel in storage.

d) Location: RB = reactor building, AB = auxiliary building, grd = pool at grade level, ele = pool at high elevation in building.

e) Seismic design basis as a function of the gravitational acceleration (g): DBE = design basis earthquake, or equivalent as used for older vintage plants; SSE = safe shutdown earthquake as defined in 10 CFR 100, App. A. Entry shown is the horizontal component.

f) Brunswick-1 has in storage 160 PWR + 656 BWR assemblies, equivalent to 1056 BWR assemblies.

g) Brunswick-2 has in storage 144 PWR + 564 BWR assemblies, equivalent to 924 BWR assemblies.

h) Dresden Units 2 and 3 have two pools in one structure. The data cited are total of the two.

i) N/A = data not available.

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Table 1.1 (Cont'd) PWR's: DATA ON SPENT FUEL STORAGE BASINS. Included are spent fuel storage inventories as of December 1984, fractions of core in storage, comparisons with the "reference case" of radionuclide inventory, locations of spent fuel basins, and seismic design bases of pools.

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[a]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Arkansas-1	2568	177	388	2.19	56.3	AB, grd	DBE=0.2g
Arkansas-2	2815	177	168	0.95	26.7	AB, grd	DBE=0.2g
Beaver Valley-1	2660	157	104	0.66	17.6	FB, grd	SSE=0.125g
Byron-1	N/A[f]	N/A	0	0.00	0.0	AB, grd	SSE=0.2g
Callaway-1	3411	N/A	N/A	N/A	N/A	AB, grd	SSE=0.2g
Calvert Cliffs-1	2700	217	868[g]	4.00[g]	108.0[g]	AB, grd	DBE=0.15g
Calvert Cliffs-2	2700	217	-	-	-	AB, grd	DBE=0.15g
Catawba-1	N/A	N/A	N/A	N/A	N/A	AB, grd	SSE=0.12g
Cook-1	3250	193	553[g]	2.87[g]	93.1[g]	AB, grd	SSE=0.20g
Cook-2	3411	193	-	-	-	AB, grd	SSE=0.20g
Crystal River-3	2544	177	171	0.97	24.6	AB, grd	SSE=0.10g
Davis Besse-1	2772	177	199	1.12	31.2	AB, grd	DBE=0.15g
Diablo Canyon-1	3338	N/A	N/A	N/A	N/A	AB, grd	DDE=0.4g[j]
Farley-1	2652	157	114	0.73	19.3	AB, grd	SSE=0.10g

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Table 1.1 (Cont'd)

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[a]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Farley-2	2652	157	62	0.39	10.5	AB, grd	SSE=0.10g
Fort Calhoun	1500	133	305	2.29	34.4	AB, grd	DBE=0.17g
Ginna	1520	121	340	2.81	42.7	AB, grd	DBE=0.20g
Haddam Neck	1825	157	545	3.47	63.4	AB, grd	DBE=0.17g
Indian Point-1	h	0[h]	160	h	h	AB, grd	DBE=0.10g
Indian Point-2	2758	193	332	1.72	47.4	AB, grd	DBE=0.15g
Indian Point-3	3025	193	140	0.73	21.9	AB, grd	DBE=0.15g
Kewaunee	1650	121	268	2.21	36.5	AB, grd	DBE=0.12g
Maine Yankee	2630	217	577	2.66	69.9	AB, grd	DBE=0.10g
McGuire-1	3411	193	91	0.47	16.1	AB, grd	SSE=0.15g
McGuire-2	3411	N/A	N/A	N/A	N/A	AB, grd	SSE=0.15g
Millstone-2	2700	217	376	1.73	46.8	AB, grd	DBE=0.17g
North Anna-1	2775	157	220[g]	1.40[g]	38.9[g]	AB, grd	SSE=0.12g
North Anna-2	2775	157	-	-	-	AB, grd	SSE=0.12g
Oconee-1	2568	177	1037[g]	5.86[g]	160.5[g]	AB, grd	DBE=0.10g

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Table 1.1 (Cont'd)

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[a]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Oconee-2	2568	177	-	-	-	AB, grd	DBE=0.1g
Oconee-3	2568	177	218	1.23	31.6	AB, grd	DBE=0.1g
Palisades	2530	204	480	2.35	59.5	AB, grd	DBE=0.20g
Palo Verde-1	N/A	N/A	N/A	N/A	N/A	AB, grd	SSE=0.20g
Point Beach-1	1518	121	524[g]	4.33[g]	65.7[g]	AB, grd	DBE=0.18g
Point Beach-2	1518	121	-	-	-	AB, grd	DBE=0.18g
Prairie Island-1	1650	121	601[g]	4.97[g]	82.0[g]	AB, grd	SSE=0.12g
Prairie Island-2	1650	121	-	-	-	AB, grd	SSE=0.12g
Rancho Seco-1	2772	177	260	1.47	40.7	AB, grd	SSE=0.25g
Robinson-2	2300	157	152	0.97	22.3	AB, grd	DBE=0.20g
Salem-1	3338	193	296	1.53	51.2	AB, grd	DBE=0.20g
Salem-2	3411	193	265	1.37	46.8	AB, grd	DBE=0.20g
San Onofre-1	1347	157	94	0.60	8.1	AB, grd	DBE=0.50g
San Onofre-2	3410	217	217	1.00	34.1	AB, grd	SSE=0.67g
San Onofre-3	3390	217	0	0.00	0.0	AB, grd	SSE=0.67g

000233

Table 1.1 (Cont'd)

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[a]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Sequoyah-1	3411	193	65	0.34	11.5	AB, grd	SSE=0.18g
Sequoyah-2	3411	193	130	0.67	23.0	AB, grd	SSE=0.18g
St. Lucie-1	2700	217	352	1.62	43.8	AB, grd	DBE=0.10g
St. Lucie-2	2560	N/A	N/A	N/A	N/A	AB, grd	SSE=0.10g
Summer-1	2775	157	52	0.33	9.2	AB, grd	SSE=0.15g
Surry-1	2441	157	608[g]	3.87[g]	94.6[g]	AB, grd	SSE=0.15g
Surry-2	2441	157	-	-	-	AB, grd	SSE=0.15g
Three Mile Island-1	2535	177	208	1.18	29.8	AB, grd	DBE=0.12g
Three Mile Island-2	1	177	0	0.00	0.0	AB, grd	SSE=0.12g
Trojan	3411	193	312	1.62	55.1	AB, grd	DBE=0.25g
Turkey Point-3	2200	157	445	2.83	62.4	AB, grd	DBE=0.15g
Turkey Point-4	2200	157	430	2.74	60.3	AB, grd	DBE=0.15g
Waterford-3	N/A	N/A	N/A	N/A	N/A	AB, grd	SSE=0.10g
Yankee Rowe	600	76	250	3.29	19.7	AB, grd	None

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Table 1.1 (Cont'd)

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[a]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Zion-1	3250	193	863[g]	4.47[g]	145.3[g]	AB, grd	SSE=0.17g
Zion-2	3250	193	-	-	-	AB, grd	SSE=0.17g

Footnotes

a) Source: U. S. Nuclear Regulatory Commission, Licensed Operating Reactors, NUREG-0020, Vol. 9, No. 1, January 1985.

b) (Stored Assemblies)/(Assemblies in Core).

c) "Reference Source Term" assumes a thermal power of 3000 MWt, stored inventory from ten annual discharges, last discharge six months ago, total inventory 700 assemblies. Source term relative to "Reference Source Term" has not been corrected for age of fuel in storage.

d) Location: RB = reactor building, AB = auxiliary building, FB = fuel building, g = pool at grade level, e = pool at high elevation in building.

e) Seismic design basis as a fraction of the gravitational acceleration (g): DBE = design basis earthquake, or equivalent as used for older vintage plants; SSE = safe shutdown earthquake as defined in 10 CFR 100, App. A. Entry shown is the horizontal component.

f) N/A = data not available.

g) Spent fuel basin shared by two units. Entries shown are totals.

h) Indian Point-1 is permanently shutdown.

i) TMI-2 is indefinitely shutdown.

j) Diablo Canyon originally used the "Double Design Earthquake," DDE acceleration = 2 DBE. Later, more elaborate analysis was done to postulate an earthquake of 0.5g associated with the Hosgri Fault.

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2. ACCIDENT INITIATING EVENTS AND PROBABILITY ESTIMATES

2.1 Loss of Water Circulating Capability

The spent fuel basins of U.S. nuclear power stations contain a large inventory of water, primarily to provide ample radiation shielding over the top of the stored spent fuel. Some typical pool dimensions and water inventories are shown in Table 2.1. The heat load from decay heat of spent fuel depends on decay time since the last refueling. Heat loads for the entire spent fuel inventory of the two older plants are shown in Table 2.2 (data extrapolated to the 1987 scheduled refuelings). The cooling systems provided for spent fuel pools typically have a capacity in the range of 15 to 20×10^6 Btu/hr (4.4 to 5.9×10^3 kw).

In the event that normal circulation of the cooling water is disrupted, e.g., due to station blackout, pump failure, pipe rupture, etc., the water temperature of the pool would steadily increase until bulk boiling occurred. (Note: In a situation where the stored inventory was small, an equilibrium temperature, below the boiling point, would be reached at which surface evaporation balanced the decay heat load).

Thermal-hydraulic analyses of the consequences of partial or complete loss of pool cooling capability are a routine part of the safety analysis reports required for licensing and amendments thereto. Generally, these analyses consider several scenarios ranging from typical to extremely conservative conditions. A sampling of conservative results for several plants is given in Table 2.3. The data clearly demonstrate that the time interval from loss of circulation until exposure of fuel to air is quite long. Even in the most pessimistic case cited in Table 2.3 (Docket No. 50-247), the water level in the pool would drop only about 6 inches per hour. Thus, there is considerable time available to restore normal cooling or to implement one of several alternative backup options for cooling.

For licensing purposes, it has been accepted that the time interval for restoring cooling manually from available water sources is adequate without requiring active (automatic) redundant cooling systems.

However, in considering the prioritization of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools," the NRC staff recognized that there is a finite probability that cooling could not be restored in a timely manner.[2] The case treated in Ref. 2 was for a BWR. The estimated frequency for the loss of one (of two) cooling "trains" was taken to be 0.1/Ry (the value assumed in WASH-1400).[3] This combined with the conditional probabilities of failure/non-availability of the second "train" yielded a combined frequency of a pool heatup event of 3.7×10^{-2}/Ry. (This estimate appears to be somewhat conservative since no "pool heatup events" are on record after $\sim 10^3$ reactor years of accumulated experience).

To escalate from a "pool heatup event" to an event which results in fuel damage requires the failure of several alternative systems that are capable of supplying makeup water (the RHR and condensate transfer systems, or, as a last resort, a fire hose). Estimated frequencies of failure for each of the alternatives, combined with the frequency of a pool heatup event, resulted in an

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estimated frequency of 1.4×10^{-6}/Ry for an accident initiated by loss of spent fuel pool cooling.

Originally, the spent fuel pool at the Ginna plant had only one installed cooling train with a "skid-mounted" backup pump and heat exchanger. However, a second cooling train was to have been installed in 1986.[4] Because of the third option for cooling at Ginna (the skid-mounted system) the probability estimate for an accident initiated by a pool heatup event should be reduced to 5×10^{-7}/Ry, i.e., about a factor of 3 smaller than for the BWR case analyzed in Ref. 2. For other PWRs with a typical two train pool cooling system, a somewhat higher failure frequency (about 10^{-6}/Ry) would be expected.

2.2 Structural Failure of Pool

Because of the massive reinforced concrete structure of LWR spent fuel storage pools, designed to Category I seismic criteria, initiating events that would lead to a structural failure are extremely unlikely. On the other hand, a structural failure that resulted in rapid and complete draining of water from the pool would have serious consequences. Probabilities of events that might result in loss of structural integrity are estimated in the following two subsections.

2.2.1 Structural Failure of Pool Resulting from Seismic Events

Procedures and conventions for a detailed probabilistic risk assessment (PRA) of seismically-induced core damage accident sequences have been presented in Ref. 5. The recommended methodology could be applied to spent fuel pools as a separate plant component, or could be coupled to a core damage sequence that might occur simultaneously during a severe earthquake. To date the seismic PRA methodology has not been rigorously applied to spent fuel pools.

Seismic risk analyses consist of three basic steps:

1) portrayal of the seismic hazard in terms of annual frequency of exceedance as a function of some ground motion parameter (e.g., the peak ground acceleration);

2) assessment of the probability that the capacity of a structure or component can survive the seismic event, often expressed in the form of a fragility curve which is the inverse of the capacity for survival; and, finally,

3) a logic model, e.g., an event tree, which relates a seismic-induced failure to a higher order event that results in some category of radioactive release.

In principle, an appropriate convolution of the probability functions derived in steps 1) and 2) yields a probability function for seismic-induced failure. It is recognized that large uncertainties exist in the two input probability functions which are reflected in the function expressing the probability of failure.

The three steps and the treatment of the uncertainties have been summarized by Reed,[6] who notes that the largest uncertainties are associated with step 1), i.e., the probabilities of occurrence of severe earthquakes having correspondingly very large ground accelerations. Reed makes the assertion that "due to the large uncertainties in the ground shaking hazard, it is unproductive to refine the structure and equipment capacity calculations to accuracies which are inconsistent with the hazard uncertainty."[6] The specific applicability to spent fuel pools of Reed's assertion is discussed in Section 2.2.1.3.

2.2.1.1 A Review of Seismic Hazard Data

The primary difficulty in characterizing the seismic hazard at specific sites in the Eastern United States (EUS), i.e., sites to the east of the Rocky Mountains is that severe earthquakes are rare events in the EUS. A systematic analysis of recorded earthquakes and their relationship to geological features has yielded seismic zonation maps of the EUS.[7] However, such information cannot readily be translated into the type of seismic hazard functions needed as input for PRA. Consequently, available historical data alone are insufficient for obtaining meaningful site specific estimates of the frequency of severe events.

During the past 6 or 7 years, the methodologies for seismic hazard analyses have been under intensive development. Hence, the analyses presented in this report must be considered provisional and subject to future refinement. At the present time, an intensive effort to refine the methodology is in progress under the auspices of the Electric Power Research Institute (EPRI).[8] The methods, input parameters, computer programming and users' manuals are presented in a ten volume report which is currently in the process of distribution.[8] This is referred to as the Seismicity Owners Group (SOG) seismic hazard methodology development program, or SOG Methodology. Unfortunately the SOG Methodology was not available for the calculations carried out in this report.

The SOG Methodology is a refinement and elaboration of the methodologies developed earlier at Lawrence Livermore National Laboratory (LLNL) by D.L. Bernreuter and his colleagues under NRC sponsorship. The initial study was a part of the NRC's Systematic Evaluation Program (SEP).[9] The methodology has been expanded and modified in a subsequent study, "EUS Seismic Hazard Characterization Project" (SHCP).[10,11]

Since the SHCP results are used for the seismic hazard estimates, some further discussion of the Bernreuter methodology is appropriate. Three basic steps are involved:

1. Expert opinion was elicited to delineate and characterize seismically active zones in the EUS, and to define earthquake ground motion models. The experts also provided estimates of uncertainties associated with their assumptions.

2. Seismic zonation, seismicity and ground motion inputs are integrated into hazard functions at specified sites.

3. Modeling and parameter uncertainties are reflected in the form of "best estimates" and 15th, 50th and 85th percentile seismic hazard curves.

The various steps are carried out in a highly disciplined and systematic manner. Provision is made at various stages for peer review of the methods and input opinion, feedback to the experts and critical evaluation of the results.

In step 1, each expert prepares a "best estimate" map which delineates the seismic zones. Each zone is characterized by a set of parameters that give the maximum earthquake intensity to be expected for that zone (upper magnitude cut-off), the expected frequency of earthquakes, and the magnitude recurrence relation. For each input (zone boundaries, seismic parameters), the expert provides a measure of his degree of confidence. Also each expert is given the option of submitting alternative maps of differing zonations and characterizations (up to as many as 30 maps). The data from each expert are evaluated separately through step 2.

In step 2, the contribution at a given site from each zone is integrated over the zone area and then over all zones. This requires the use of ground motion models for which a range of alternative models are employed to yield a set of alternative hazard curves. A "Ground Motion Panel" of experts have selected several alternative models to be used, each having a weighting factor (see Ref. 10, App. C). Also each ground motion model incorporates a site specific correction to account for local geology.

In step 3, the results of the individual experts are combined to obtain a "best estimate" hazard curve and the uncertainty bands are computed in several alternative ways.

It is obvious that the methodology requires a massive data collection and computer effort. In its present state, the final results are not in a form to be easily applied to a specific PRA by a non-expert in seismology. Further work is needed to develop a more convenient format for presenting the final results. In particular, numerical tabulations of the sets of hazard curves (such as those shown in Figs. 2.1 and 2.2) and their derivatives, $|dH/da|_i$, for each reactor site would be helpful. Also, it appears that the local site geology needs more rigorous consideration in the derivation of the hazard curves (see below).

Members of the Peer Review Panel have suggested several ways in which the methodology could be refined (see Ref. 11, Section 7 and Appendices D.1-D.4). Many of these suggestions were implemented in the final feedback process and were included in the final results reported in Ref. 11.

In order to illustrate the hazard curves, their range of uncertainties and comparison with other studies, a series of figures taken from Ref. 11 for the Millstone site is reproduced in Figs. 2.1-2.4.

Figure 2.1 is the hazard curve obtained from averaging the "best estimate" results for all experts in the SHCP study (including the seismic and the ground motions panels). The curve plots frequency of exceedance per year vs. peak ground acceleration.

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Figure 2.2 illustrates the uncertainties in the hazard curve (15, 50, and 85 percentiles) derived from the spread in expert opinion and the self-confidence factors in the input parameters. It can be seen that the spread between the 15 and 85 percentiles is about a factor of 20 at low PGA increasing to about 50 at the high PGA. Comparison of Figs. 2.1 and 2.2 shows that the "best estimate" curve is considerably higher than the 50 percentile, i.e., the mean > median.

Figure 2.3 illustrates the spread in the "best estimate" hazard curves for all of the experts participating in either the SEP[9] or the SHCP[10] studies, or both (6 experts participated in both studies). The spread ranges from about one order of magnitude at lower PGA to about 1.5 orders of magnitude at the higher PGA. The curve marked "A," which falls considerably below the main grouping, was derived from data input in the SEP study by one of the experts who participated in both studies. The revised input for this expert in the SHCP project raised the derived curve by about an order of magnitude at the low accelerations and almost two orders of magnitude at the higher PGA. This raises the obvious questions of whether the experts were somehow influenced by the opinions of their colleagues, or whether the revision resulted from a more careful consideration of the various geological factors that were taken into account in preparing the input parameters. The question of testing the results for inadvertent biases of this nature was addressed by the Peer Review Panel members, but their recommendations could not be fully implemented in the final report due to limited time and budget (Ref. 11, pg. 7-3).

Figure 2.4 compares the "best estimate" hazard curves for the individual SHCP experts with curves generated from zonation maps prepared by the U.S. Geological Survey (USGS)[12] and historical data of the past 280 years. As can be seen, the USGS hazard curve (denoted by "X") lies above the SHCP data. Bernreuter et al. attribute the difference between the SHCP and the USGS curves to the variations in the equations used for conversions from intensity to magnitude and in the values for the rate of earthquake recurrence (Ref. 11, pg. 8-1 et seq.). As would be expected the 280 year historical hazard curve (denoted by "H") falls below the SHCP data because it does not include postulated stronger earthquakes with return times much greater than the time span of the historical record.

It should be noted that recent research has raised significant questions concerning the frequency of strong earthquakes in the coastal zone of the EUS.[13] The speculation has arisen from paleoseismic field studies originally focused on the region of the strong earthquake near Charleston, SC, in 1886, which produced many "sand blows."[14,15] These result from the liquefaction and venting to the surface of sub-surface water-saturated sediment. Several sand blow craters have been found for which radiocarbon dating indicates that moderate to large earthquakes have recurred in the Charleston region on an average of about every 1800 years.[16] The latest (prior to 1886) occurred about 1100 years ago.[16] Sand blows from prehistoric earthquakes have been unearthed recently in the region extending from near Savannah, GA as far north as Myrtle Beach, SC.[17] The broad extent of sand blows suggests that Charleston-type earthquakes might be associated with some tectonic feature which extends for some distance along the east coast and not uniquely centered near Charleston. Up to the present time, no systematic field search has been made for sand blows outside of the Savannah to Myrtle Beach region.[18] Recently Thorson et al. reported the existence of apparent sand blow craters in eastern

Connecticut.[19] These craters were recently examined by a USGS field team and assessed as not being of the same nature as those observed in South Carolina.[18]

2.2.1.2 Seismic Hazard Estimates for Eastern United States Sites

The "best estimate" and the median, 15 and 85 percentile seismic hazard curves developed by the SHCP project for the Millstone site are shown in Figs. 2.1 and 2.2.[11] An NRC staff evaluation of the seismic hazard curves for the Millstone site was performed as part of the Millstone Risk Evaluation[20] (Appendix D). Included in the staff evaluation were discussions of seismic hazard curves generated by contractors of the Licensee and used in the Millstone 3 Probabilistic Safety Study (PSS). The PSS 16, 50 and 84 fractile hazard curves are shown in Fig. D.1 of Ref. 20 together with the 15, 50 and 85 fractile curves from the initial phase of the SHCP project.[10] The staff noted that the PSS hazard curves were about an order of magnitude below the SHCP curves. However, the final revisions of the SHCP data[11] decreased the discrepancy by more than a factor of two. The revised SHCP curves are compared with the PSS 50 fractile curve in Fig. 2.5. It will be seen that the PSS 50 fractile curve almost coincides with the SHCP 15 fractile.

Although the hazard curves shown in Figs. 2.1 through 2.5 are intended to be "site specific" for the Millstone site, comparison with curves generated for other EUS sites, generally fall within the range of uncertainty reflected by the spread in Fig. 2.5. Therefore, the spread of these seismic hazard curves can be regarded as "generic" rather than site specific.

The data in the four SHCP hazard curves were used for the calculations presented in Section 2.2.1.4. The PSS 50 fractile curve was included in the sensitivity cases presented in Section 2.2.1.5.

A fundamental problem in attempting to use the hazard curves is the need for extrapolation to higher peak ground accelerations, e.g., to values as high as 2000 cm/sec^2 (~2g). The NRC staff evaluation[20] recognizes the validity of choosing an upper magnitude cutoff for each seismic zone but underscores the uncertainty in such parameters (see Ref. 20, pg. D-3). The choice of the upper magnitude cutoffs is one of several factors that determine how fast the tails of the hazard curves drop at peak ground accelerations in excess of about 600-700 cm/sec^2. In order to illustrate how the seismic failure probabilities might be affected, we have extrapolated the four SHCP curves to >2g using an exponential tail tangent to the point where the hazard curves terminate in Figs. 2.1 and 2.2. The results are presented in Section 2.2.1.5.

2.2.1.3 Seismic Fragility of Pool Structures

Fragility curves specifically for spent fuel pools have never been developed.[21] It is necessary therefore, to rely on fragility assessments for other structures which appear to be of similar construction to spent fuel storage pools. It must be recognized that this procedure introduces an additional element of uncertainty in the final risk estimates -- an uncertainty that is difficult to quantify. Another source of uncertainty is the degree to which the stainless steel lining of a pool would enhance the seismic strength capacity (i.e., reduce the fragility). Conceivably, the reinforced concrete structure of the pool could crack without loss of integrity of the pool lining.

The dilemma of selecting an appropriate fragility for a BWR plant is aggravated by the fact that the pool structure extends typically from the 60 to the 100 foot elevations above grade with the resultant amplification of the seismic bending stresses relative to the lower elevations of the structure.[22]

For the present analyses, two, somewhat diverse sets of fragility estimates, have been used:

1) the fragility curve developed by R.P. Kennedy et al.[23] for the Oyster Creek reactor building; and

2) the fragility of the Zion plant auxiliary building shear walls (north-south ground motion).[24]

For the sensitivity studies presented in Section 2.2.1.5, a third fragility curve has been added which is for a 36 inch thick reinforced concrete shear wall.[25]

In each case, the fragility curve is defined by the following equation:

$$F(a) = \Phi\,[\ln(a/\check{A})/\beta_R]\ , \qquad (2.1)$$

where F(a) is the probability of structural failure given a peak ground acceleration, PGA = a. $\Phi(\ )$ is the normal distribution function, $\check{A}$ is the median fragility level (i.e., the acceleration at which there is a 50% probability of failure) and β_R is the logarithmic standard deviation expressing the randomness in the value of $\check{A}$. A third parameter, β_U, is used to express the uncertainty in the median value and is used to generate upper and lower confidence limits. For example, it can be shown that the substitution for $\check{A}$ in Eq. 2.1 of $\check{A}' = \check{A}\, e^{+\beta_U}$ and $\check{A}' = e^{-\beta_U}$ generate respectively the 84 and 16 percentile curves.

Thus, a set of fragility curves can be generated from three parameters, $\check{A}$, β_R and β_U. The data used for generating the "Kennedy" and the "Zion" curves are given in Table 2.4.

Kennedy notes that the estimated median fragility value of about 0.75g is considered applicable to plants designed in the U.S. in the mid 1960's. The Kennedy fragility curve is shown in Fig. 2.6, with the 84 and 16 percentile limits. The corresponding Zion curves appear in Fig. 22, pp. 3-35 of Ref. 24. (Note: the Zion curves are expressed in terms of floor slab acceleration, rather than ground acceleration.)

It has been noted by Kennedy et al.[23] that the

"use of the lognormal distribution for estimating very low failure frequencies of components or structures associated with the tails of the distribution is considered to be conservative since the low probability tails of the lognormal distribution generally extend further from the median than actual structural response data might extend since such data generally show cut-off limits beyond which there is essentially zero probability of occurrence."

In other words, Kennedy indicates that a more realistic analysis should truncate the lower end of the fragility curve as indicated by experimental data. (We chose to truncate when $F(a) < 10^{-4}$ in Equation 2.1).

2.2.1.4 Seismically-Induced Failure Probabilities

The convolution of the derivative of a seismic hazard curve (e.g., Fig. 2.1) with a fragility curve, yields the annual probability of a seismically-induced failure. This can be expressed by the equation:

$$P_{i,j} = \int_0^{a_{max}} \left|\frac{dH}{da}\right|_i F(a)_j \, da \, , \qquad (2.2)$$

where $P_{i,j}$ is the failure probability obtained from the convolution of hazard curve i with fragility curve j, $|dH/da|_i$ is the derivative of the hazard curve i (i.e., the annual frequency of occurrence of peak ground acceleration, a, and $F(a)_j$ is failure probability at acceleration, a, for fragility curve j. Since the seismic hazard curve is not an analytic function, the derivative dH/da and the integration are carried out numerically.

Given many hazard and fragility curves from which to choose, and there being no a priori basis for choosing a particular pair, the convolution expressed in Eq. 2.2 can be carried out for each pair of curves with weighting factors assigned to each of the curves in each set. The resultant collection of $P_{i,j}$ gives a probability distribution which expresses the uncertainties in the analysis. The probability density distribution obtained for the generic site is shown in Fig. 2.7.

At least in principle, the various hazard and fragility curves (sets i and j) do not have an equal likelihood of being correct. Therefore, a weighting factor (ω_i or ω_j) should be assigned to each curve which reflects an "engineering judgement" of its relative validity for a specific site. The mean probability for failure is then derived from the following expression,

$$\bar{P}_f = \sum \omega_i \omega_j P_{i,j} \, / \, \sum \omega_{i,j} \, , \qquad (2.3)$$

where $\sum \omega_i = 1$, $\sum \omega_j = 1$ and $\sum \omega_{i,j} = \sum \omega_i \omega_j = 1$. For the purposes of illustration, the weighting factors listed in Table 2.5 were selected. Results for each convolution without weighting factors are listed in Table 2.6. As can be seen from the table, the "best estimate" hazard curve has been assigned a weighting factor of 0.5 with the remaining 0.5 distributed among the median, 15 and 85 percentile curves. The "Kennedy" set of fragility curves were assigned a total weighting factor of 0.75 with the remaining 0.25 distributed among the "Zion" set. Assuming an upper limit cutoff of 1.0 g, the mean probability of failure, $\bar{P}_f$, derived from the 24 sets of $P_{i,j}$, using the weighting factors listed in Table 2.5 and Equation 2.3, would be

$$\bar{P}_f = 2.2 \times 10^{-5}/\text{year} \, .$$

2.2.1.5 Sensitivity Studies

In order to better understand the limitations in assessing the probability of seismically-induced structural failures in spent fuel pools the results of the convolutions of each pair of curves (seismic hazard against fragility) are listed in Table 2.6. Each of the five hazard curves were convoluted with seven fragility curves. Results range from

$$P_f = 2.6\times10^{-4} \text{ to } \ll 1\times10^{-10}/\text{Ry} .$$

The primary contributors to the wide spread in the results are:

- uncertainties in how the tails of seismic hazard curves drop off at PGA's < 700 cm/sec^2.
- uncertainties in pool structural fragilities.

2.2.1.6 Conclusions on Seismic Risk

Available seismic hazard and fragility data are inadequate to assess the probability of seismically-induced structural failure of spent fuel pools within a factor of 1000. The present results indicate that the pool failure probability may be as high as 2.6×10^{-4}/Ry, or as low as 1×10^{-10}/Ry. Further work is need to narrow the uncertainties.

2.2.2 Structural Failures of Pool Due to Missiles

Missiles generated by tornadoes, aircraft crashes or turbine failure could penetrate the pool structure and result in structural failure.

The probability of tornado missiles depends on the frequency of tornadoes at the site, the target area presented to the missile and the angle of impact. An analysis made by Orvis et al.[26] for an average U.S. site derives a probability of $<1\times10^{-8}$/year for structural loss of pool integrity due to a tornado missile (Ref. 26, pg. 4-44). (WASH-1400 estimated $<5\times10^{-6}$/yr.[3])

Similarly, the analysis for structural failure of a pool from an aircraft crash yielded a probability of $<1\times10^{-10}$/year (Ref. 26, pg. 4-58).

The damage caused by Missiles generated by turbine failure depends on the orientation of the turbine axis relative to the structure, as well as the frequency of turbine failure. An analysis by Bush yields a probability of $\sim4\times10^{-7}$/year for spent fuel pool damage from a turbine failure missile.[3,27] In the case of Ginna, the probability would be several orders of magnitude smaller (i.e., essentially zero) because the spent fuel pool is shielded from turbine missiles by the primary containment.

2.3 Partial Draindown of Pool Due to Refueling Cavity Seal Failures

On August 21, 1984, the Haddam Neck Plant experienced a failure of the refueling cavity water seal, while the refueling cavity was flooded. The water level in the refueling cavity dropped by about 23 feet to the top of the reactor vessel flange within 20 minutes -- a loss of approximately 200,000 gallons, or a leak rate of about 10,000 gallons per minute.[28] At the time of

the event, refueling had not begun. The gates of the transfer tube connecting the refueling cavity to the spent fuel storage pool were closed.

Although the seal failure did not result in an accident or in the release of radioactivity, the incident raised the question of whether similar failures might occur while spent fuel was being transferred or while transfer gates to the spent fuel basin were open, either case of which might result in exposure of spent fuel to air and possible clad failure.

All licensed plants were instructed to evaluate the potential for and consequences of a refueling cavity seal failure.[28]

Refueling cavity seals seal the gap between the reactor vessel flange and a flange on the inner periphery of the reactor cavity, or the floor of the cavity.

Most BWR's have a permanently installed stainless steel bellows to seal the gap, and are thus not subject to failure of the Haddam Neck type.

Many PWR's seal the gap with gaskets held down by a bolted flat steel ring. Such systems have experienced difficulties in achieving tight seal because of surface irregularities and small vertical and concentric offsets in the two flanges. Consequently, many plants have converted to inflatable (pneumatic) rubber seals. Also, it should be noted that pneumatic rubber seals are often used to seal the gates in transfer tubes or canals.

Licensee responses to the IE Bulletin indicate that the Haddam Neck cavity configuration is unique in that the width of the annular gap between the reactor flange and the cavity flange is about two feet, whereas, in most plants the gap is of the order of <1" to -3". As of summer 1985 some 45 units used pneumatic seals in the refueling cavity.[29]

Typical pneumatic seals are illustrated in Figures 2.8-2.10. There are many variations in the details of the designs, e.g., some plants have various types of retainers to support the rubber seals (e.g., see Figure 2.10), others rely on the rubber seal alone (e.g., see Figure 2.9). According to the responses of the licensees, even if a pneumatic seal should deflate, the leakage would be expected to be small or negligible, because the wedged shaped upper section would maintain a good seal (refer to Figure 2.8), i.e., the deflated seal would not distort enough under the hydrostatic head to extrude through the gap.

Aside from the Haddam Neck 1984 incident, a few cases have been reported in which inflated seals have failed, either in the refueling cavity or transfer gates. None of these events had significant radiological consequences. The most serious loss of pool water inventory occurred December 3, 1986 at the Hatch 1 & 2 spent fuel pool when a pair of pneumatic seals in the seismic gap of the transfer canal deflated. About 141,000 gallons of water was lost from the pool and the pool level dropped about five and one half feet before the leak was detected.[30]

Several seal failure events are listed in Table 2.7. It is likely that this list is not exhaustive. To the best of the authors' knowledge no data base has been compiled (or is available) of the failure rate of pneumatic

seals and their pressurizing systems of the types used in nuclear power plants, or of similar seals used in non-nuclear industries.

Based on the limited experience cited in Table 2.7, the historical failure rate in seals/systems is in the range of $\sim 1 \times 10^{-2}$/Ry. Because of advances in design, increased awareness and surveillance, the present failure rate is estimated to be an order of magnitude smaller, i.e., $\sim 1 \times 10^{-3}$/Ry.

As is obvious from Table 2.7, a seal failure does not necessarily result in the rapid loss of water inventory from spent fuel transit or storage locations. The limited experience indicates that the most probable time for a refueling cavity seal to fail is shortly after installation, while the cavity volume is being filled with water. According to the analyses supplied by licensees in response to IE Bulletin No. 84-03, the failure of a pneumatic refueling cavity seal in most PWR plants would not result in massive leaks because of the relatively narrow gap to be sealed and the geometric shape of the seal. Also, leaks from seal failures in transfer tube/canal gates would be limited, in most cases, because the leakage would be into a confined volume, e.g., from the storage pool into a drained up-ender sump. Taking these factors into consideration, it is estimated that the frequency of a serious loss of pool water inventory resulting from a pneumatic seal failure to be in the range of $\sim 1 \times 10^{-5}$/Ry.

Even a large loss of water inventory from the spent fuel pool does not necessarily result in uncovering and subsequent failure of fuel. Most spent fuel pools are constructed with weirs below the transfer gates which preclude complete drainage of the pool, even in the event of a Haddam Neck type failure with the transfer tube/canal gates open. In most cases, the water level would remain a foot or more above the active zone of the spent fuel assemblies. In a few cases, the upper several inches of the fuel could uncover. (Note: Licensee responses to IE Bulletin 84-03 did not always provide information about the elevations of weirs and tops of stored assemblies.)

In the event of a draindown of the pool to near the top of the fuel assemblies, there would still be time (1/2 to 1 hour) to close gates and restore a supply of water before the residual water inventory reached the boiling point. However, as noted in one licensee response, even if the fuel remained covered "dose rate in the vicinity of the spent fuel pool would, however, be high, complicating recovery from the event."[31]

A pool heatup event similar to the partial draindown scenario described above was considered by the NRC staff in Ref. 2. A conditional probability for failure to restore adequate makeup water was taken to be 5×10^{-2}. Combining this restoration failure frequency with the initiating frequency (1×10^{-5}/Ry), the probability of a pneumatic seal failure which results in exposure to air of stored spent fuel with possible clad failure is estimated to be of the order of

$$P = 5 \times 10^{-7}/\text{Ry}.$$

2.4 Pool Structural Failure Due to Heavy Load Drop

WASH-1400 considered the probability of structural damage to the pool due to the dropping of a fuel transfer cask (Ref. 3, pg. I-97). In the analysis,

it was anticipated that one spent fuel shipment per week would be the equilibrium shipping rate. The estimated rate for a drop resulting in pool failure (for a single unit plant) was 4.5×10^{-7}/Ry.

The above frequency was based on a crane failure probability of 3×10^{-6} per operating hour. It was further assumed that each lift was of 10 minutes duration and for a 10 second period per lift the cask would be in a position to cause gross structural damage to the pool wall if a crane failure occurred. Human error was not considered.

Since very little spent fuel is currently being shipped, the likelihood of such an accident is very low. However, at some point in the future as the fuel pools are filled, spent fuel will have to be removed from the reactor spent fuel pools, either to some onsite storage facility, or eventually to a high level waste repository. At that time, the frequency of removal of spent fuel will be correspondingly greater.

Orvis et al.[25] have reexamined the cask drop probability and have used the following probabilities:

Mechanical failure of crane = 3×10^{-6}/operating hour

Electrical control failure of crane = 3×10^{-6}/operating hour

Human error = 6×10^{-4}/lift.

As can be seen, human error dominates the Orvis estimates for probability of a cask drop. The Orvis datum for human error was based on a study by Garrick et al.[30] which concerned human reliability in the positioning of heavy objects. The applicability of the Garrick study to crane operations is not obvious. Nevertheless, a human failure rate in the range of 10^{-3} to 10^{-4} per operation appears to be consistent with data listed in the NRC handbook on human reliability analysis[31] for cases in which the operation has one or more people who serve as "checkers" and involves some degree of personal risk to the operating personnel.

Obviously, not all human failures associated with the lifting and moving of a spent fuel shipping cask would result in structural damage to the pool. The section of the pool where the cask is set down generally has an impact pad to absorb the impulse of a dropped cask. Accidents in unloading the cask from or reloading on the transport vehicle would not involve the pool.

Only horizontal movements of the cask above a structurally critical section of the pool would pose the threat of structural damage. As noted above, WASH-1400 assumed that the sensitive section is the vertical wall at the pool edge. It was implicitly assumed that <u>all</u> load drops on the pool edge would result in structural failure. This assumption appears to be too simplistic and consequently too conservative for the following reasons:

- many "load drops" would be partially attenuated by crane mechanisms which limit descent rates, and reduce impact energy,

- in case of some "off-center" hits, the full potential impact energy would not be absorbed by the pool edge (cask tilted, one end strikes floor first), and

- account should be taken of exterior cask fittings (e.g., cooling vanes) which absorb some impact energy.

No rigorous structural analyses have been performed to scope the range of damage to a pool edge from a cask drop. In the absence of such analyses, it has been necessary to estimate the conditional probability of structural damage given a cask drop in the vicinity of the pool edge. It is estimated that the conditional probability is less than 100% and greater than 1%. A conditional probability of 10% has been selected for the hazard calculation and 100% and 1% used for defining the range of uncertainties.

Since human error, rather than mechanical or electrical failure, appears to dominate the hazard arising from shipping cask movements, the various steps in the crane operation have been identified in Table 2.8, which also lists the types of human error associated with each step. The distribution of failure frequency in the various steps has been estimated and listed in the last column of Table 2.8. (This distribution was subjected to "peer review" by BNL rigging personnel and managers who oversee operations of this type[34] as well as human factors analysts.[35])

It will be noted that most steps in the crane operation do not jeopardize the structural integrity of the pool. Only in steps 5a and 5b (see Table 2.8) could the cask strike the pool edge. An accident of the type listed in 5a (horizontal movement with cask not high enough to clear the pool edge) would probably not cause serious damage because of the limited kinetic energy of the cask associated with the slow velocity of horizontal crane movements. Thus, only step 5b in Table 2.8 is considered in the hazard calculation.

For purposes of calculating the cask drop hazard, i.e., the probability of structural damage to the pool resulting from a cask dropping on the pool edge, the assumptions listed in Table 2.9 were used. Table 2.9 also lists the uncertainty ranges for each of the parameters. The results are as follows:

Probability of structural failure due to cask drop on pool edge caused by mechanical or electrical failure of crane = 3.5×10^{-7}/Ry.

Probability of structural failure due to cask drop on pool edge caused by human error = 3.1×10^{-5}/Ry.

If the failure rates summarized in Table 2.9 are assumed to be statistically independent, then the uncertainty in the overall failure rate is dominated by the uncertainty in the probability of pool failure. Thus the overall uncertainty is about a factor of ten in either direction.

The NRC has proposed[36] a number of improvements in handling of heavy loads which they estimate will substantially reduce the likelihood of load drop accidents. With the recommended improvements in procedures and equipment in place, the NRC estimated[36] that the likelihood of a cask drop or other heavy load drop over the spent fuel pool would be between 2×10^{-5} and 2×10^{-9} per reactor year. This is a substantial reduction in the likelihood of a load

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drop. With the previous estimate of pool failure per cask drop (0.001) the pool failure rate would be $2x10^{-8}$ to $2x10^{-12}$ per reactor year due to load drop accidents.

2.5 Summary of Accident Probabilities

The probability estimates made in Sections 2.1-2.4 are summarized in Table 2.10. These include only those accidents that result in the complete loss of pool water inventory. It will be seen that shipping cask drop resulting from human error and seismic induced failures dominate in the hazards. As previously discussed the uncertainty in both of these probabilities is quite large and has been estimated to be an order of magnitude in either direction.

2.6 References for Section 2

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Table 2.1 Typical Spent Fuel Pool Dimensions and Water Inventories

Length/Width/Depth (feet)	Pool Volumes (cubic feet)	Nominal Water Inventory (cubic feet)
40/26/39[a]	4.1×10^4	3.5×10^4
43/22.25/40.25[b]	3.4×10^4	3.3×10^4

[a]BWR, Vermont Yankee.
[b]PWR, Ginna.

Table 2.2 Decay Heat as a Function of Time Since Last Refueling (Data from Appendix A)

Plant	Decay Heat Load (10^6 Btu/hour) Decay Time Since Last Shutdown for Refueling			
	30 days	90 days	0.5 years	1.0 year
Millstone-1	4.43	3.10	2.38	1.76
Ginna	2.62	1.96	1.59	1.25

Table 2.3 Examples of Thermal-Hydraulic Transient Parameters, Assuming Complete Loss of Pool Coolant Circulation

Docket No.[a]	Rate of Temp. Increase (°F/hr)	Time of Boiling[b] (hours)	Boil-Off Rate (gpm)	Boil-Off Rate (ft^3/hr)
50-325	5.0	13.5	28	262
50-250	9.7	9.3	N.A.	-
50-271	<3	>20	14	131
50-247	13.0	4.8	57	534
50-344	<6.3	>11	34	318

[a]See Ref. 1.
[b]Hours after complete loss of cooling capability.

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Table 2.4 Fragility Parameters Assumed in This Study for Spent Fuel Storage Pools

Structure	A (g)	β_R	β_u	Ref.
Oyster Creek Reactor Building[a]	0.75	0.37	0.38	23
Zion Auxiliary Building Shear Walls (N-S motion)[b]	1.1	0.12	0.20	24
36-inch Thick Reinforced Concrete Shear Wall	2.15	0.22	--	25

[a]Designated as the "Kennedy" fragility curves in the text.
[b]Designated as the "Zion" fragility curves in the text.

Table 2.5 Weighting Factors Assigned to the Various Hazard and Fragility Curves for the "Weighted" Analysis

Seismic Hazard Curves:	ω_i
"Best Estimate"	0.50
15% Confidence Curve	0.10
Median Curve	0.30
85% Confidence Curve	0.10
	$\sum\omega_i$ = 1.00

Fragility Curves:	ω_j
"Kennedy", Median	0.45
" , 16%	0.15
" , 84%	0.15
"Zion", Median	0.15
" , 16%	0.05
" , 84%	0.05
	$\sum\omega_j$ = 1.00

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Table 2.6 Summary of Convolutions of Seismic Hazard Curves with Fragility Curves. The Seismically-Induced Failure Probabilties were Calculated from the Various Pairs of Curves Using Equation 2.2

Hazard Curve \ Fragility Curve →	PROBABILITY OF FAILURE PER YEAR						
	K16	K50	K84	Z16	Z50	Z84	SW36
SHCP 85	2.6E-4	8.8E-5	2.9E-5	3.0E-5	1.6E-5	6.4E-6	5.9E-7
SHCP "BE"	8.4E-5	2.9E-5	9.2E-6	1.0E-5	5.0E-6	2.0E-6	1.6E-7
SHCP 50	3.1E-5	9.3E-6	2.4E-6	2.0E-6	7.8E-7	2.4E-7	1.6E-8
SHCP 15	3.1E-6	7.0E-7	1.3E-7	6.4E-8	5.6E-8	3.5E-9	1.6E-10
PSS 50	4.8E-6	9.3E-7	1.4E-7	1.5E-8	2.9E-10	neg.[a]	neg.[a]

[a]neg. = $<10^{-10}$

Key to seismic hazard curves:

SHCP 85: Ref. 11, 85 percentile (see Fig. 2.2).
SHCP "BE": Ref. 11, "best estimate" (see Fig. 2.1).
SHCP 50: Ref. 11, 50 percentile (see Fig. 2.2).
SHCP 15: Ref. 11, 15 percentile (see Fig. 2.2).
PSS 50: Ref. 20, 50 fractile (see Fig. 2.5).

Key to fragility curves:

K16, 50, 84: Ref. 23, 16, 50 and 84 percentiles (see Fig. 2.6).
Z16, 50, 84: Ref. 24, (Zion auxiliary building shear wall), 16, 50 and 84 percentiles.
SW36: Ref. 25 (36" thick reinforced concrete shear wall).

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Table 2.7 Events in Which Inflated Seals Have Failed

Date	Plant	Seal Location	Cause	Total Leakage
9/72	Pt. Beach[1]	Transfer Gate	Failure of air supply	11,689 gal.
10/76	Brunswick 2	Inner Pool Gate	Air leak in seal plus compressor power supply failure	(Pool level dropped 5")
5/81	Arkansas Nuclear One - 2	Transfer Gate	Maintenance error, air supply shutoff	1000 gal/min
8/84	Haddam Neck	Cavity Seal	Design weakness, seal shifted	200,000 gal. in 20 min.
10/84	San Onofre 2[1]	Gate Seal	Air compressor power failure	20,000 gal.
11/84	San Onofre 2[1]	Cavity Seal[2]	Manufacturing defect, seal rupture	----
12/86	Hatch[3]	Pool-Canal Flexible Joint	Valve to compressed air supply closed	141,000 gal.

[1]No spent fuel was in the storage pool.
[2]Failure occurred during installation and leak testing.
[3]The leak went undetected for about 7-1/2 hours.

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Table 2.8 Estimated Distribution of Human Error in Heavy Crane Operations. These Estimates, Made by BNL Staff,[34,35] are Based on Engineering Judgement and are Not Supported by Actuarial Data

Operational Step	Possible Human Errors	Estimated Fraction of Total Error Frequency[a] (Per Cent)
1. Install rigging	Wrong slings (e.g., hoist rigging not qualified for task)	10
	Improper installation (shackle, pins, etc.)	10
2. Positioning of crane over load, apply tension	Crane hook not over center of gravity (load upset as tension applied)	15
3. Lift load	Control error (wrong hoisting speed, unintentional reversal of direction)	10
4. Start horizontal travel	Control error (move wrong direction, lift or lower instead of move)	10
5a. Horizontal travel load not high enough to clear obstacles	Control error (unintentional reversal of motion, overshoot stopping point)	4
5b. Horizontal travel	Control error or delayed rigging failure resulting in load drop of cask on fuel pool wall	1
6. Lower load	Control error (wrong direction, descent too fast)	10
7. Positioning of crane over receiving cradle and set-down load	Inaccurate positioning cradle capsizes during set-down	20
	Set down too rapid	10

[a]It is assumed that the movement of a spent fuel shipping cask is carried out by a qualified rigging crew consisting of a foreman, two or more riggers, and a crane operator. The foreman and riggers check each step and crane movements are signaled to the operator by the foreman who stands in a location providing adequate surveillance of the load, and can be clearly seen by the operator.

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Table 2.9 Assumptions Used in Calculating the Hazard of Catastrophic Structural Damage to Pool Resulting from the Drop of a Shipping Cask

Item	Assumed Value	Uncertainty Range
Number of fuel shipments (eventual rate to reduce accumulated inventory) per week	2	--
Number of passes over pool edge per shipment	2[a]	0
Fraction of horizontal movement when cask is above pool edge	0.25	0.1 to 0.5
Total operational time in each movement, minutes per lift	10	8 to 30
Time over pool edge per lift, seconds per lift	10	5 to 20
Mechanical failure rate of crane, per operating hour	3×10^{-6}	10^{-6} to 10^{-5}
Electrical failure rate of crane, per operating hour	3×10^{-6}	10^{-6} to 10^{-5}
Total accident rate from human error, failures per lift	6×10^{-4}	10^{-4} to 10^{-3}
Fraction of human error cask drop accidents occurring during horizontal motion of crane, fraction of total	0.01	5×10^{-3} to 5×10^{-2}
Conditional probability of structural failure of pool given a cask drop at pool edge location, failures per drop	0.1	10^{-2} to 1.0

[a]Some spent fuel pools have a special section for the shipping cask separated from the main pool by a wall with a wier or gate. For such a configuration the number of passes over the "pool edge" would be zero and hence the risk to the main pool from a cask drop would be zero.

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Table 2.10 Summary of Estimated Probabilities for Beyond Design Basis Accidents in Spent Fuel Pools Due to Complete Loss of Water Inventory

Accident	Estimated Probability/Ry	
	Millstone	Ginna
Loss of Pool Cooling Capability	1.4×10^{-6}	5.7×10^{-7}*
Seismic Structural Failure of Pool	2.6×10^{-4}-Neg.	2.6×10^{-4}-Neg.
Structural Failure from Tornado Missiles	$<1 \times 10^{-8}$	$<1 \times 10^{-8}$
Structural Failure from Aircrash	$<1 \times 10^{-10}$	$<1 \times 10^{-10}$
Structural Failure from Turbine Missile	4×10^{-7}	-0**
Loss of Pool Water Due to Pneumatic Seal Failure	0***	5×10^{-7}
Structural Failure from Cask Drop[1]	3.1×10^{-5}	3.1×10^{-5}
Structural Failure from Cask Drop after improvements recommended by Generic Issue A-36 resolution.	$<2 \times 10^{-8}$	$<2 \times 10^{-8}$

Neg. - Negligible.
[1]After removal of accumulated inventory resumes.
*With credit for third cooling system. Other PWRs which typically have two spent fuel cooling systems would have an estimated fuel uncovery frequency of about 1×10^{-6}/Ry.
**Typical PWRs may have a failure frequency due to turbine missiles on the order of 4×10^{-7} but Ginna's pool is shielded from the turbine.
***Most BWRs cannot lose pool inventory even if refueling cavity leaks.

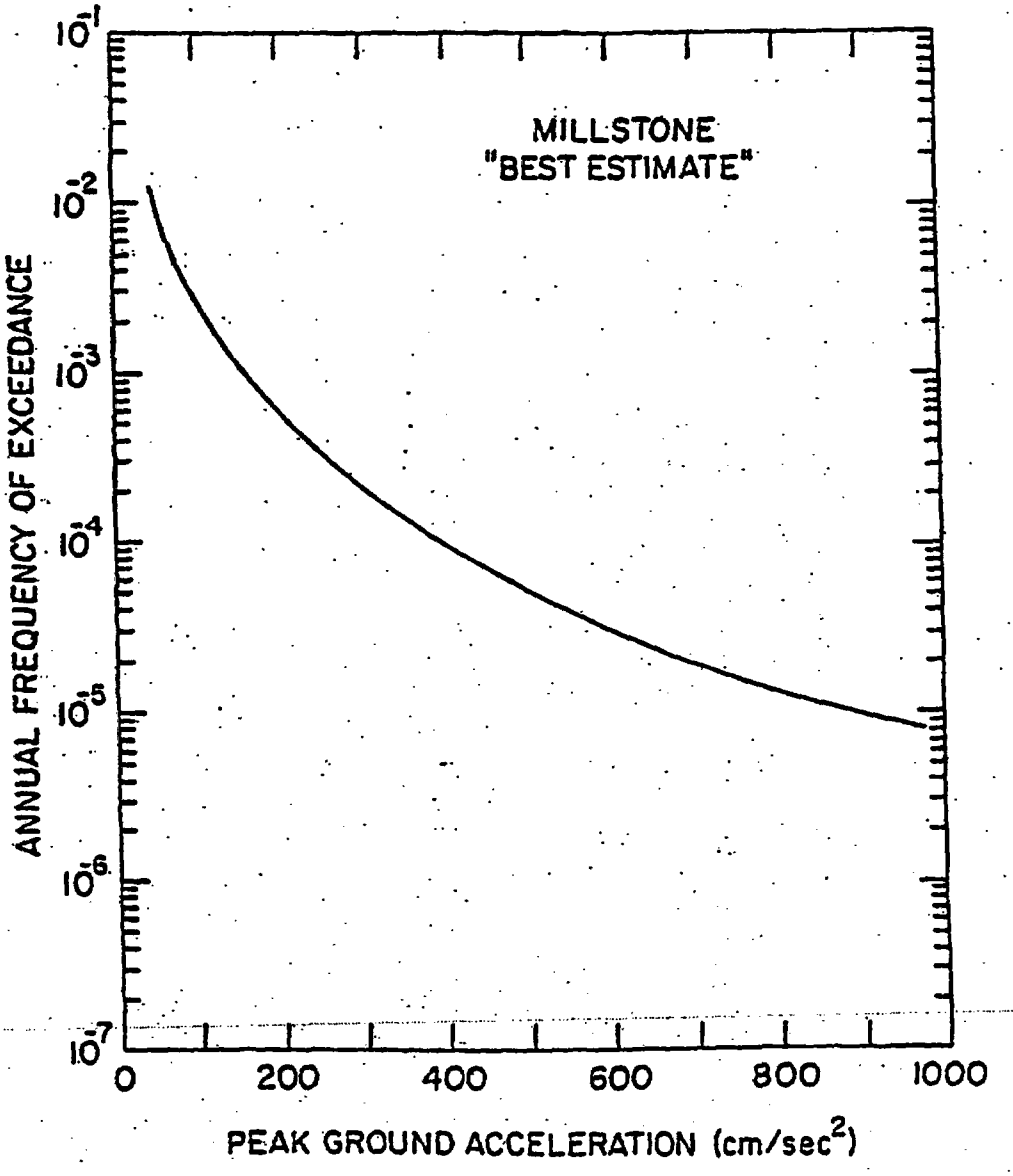

Figure 2.1 Seismic Hazard Curve for the Millstone Site. The curve shown is the mean of the hazard curves generated from the "best estimate" input data of the ten experts participating in the SHC study combined with the "best estimate" model of the ground motion panel. Site corrections are included (Source: Ref. 11, pg. 5-43).

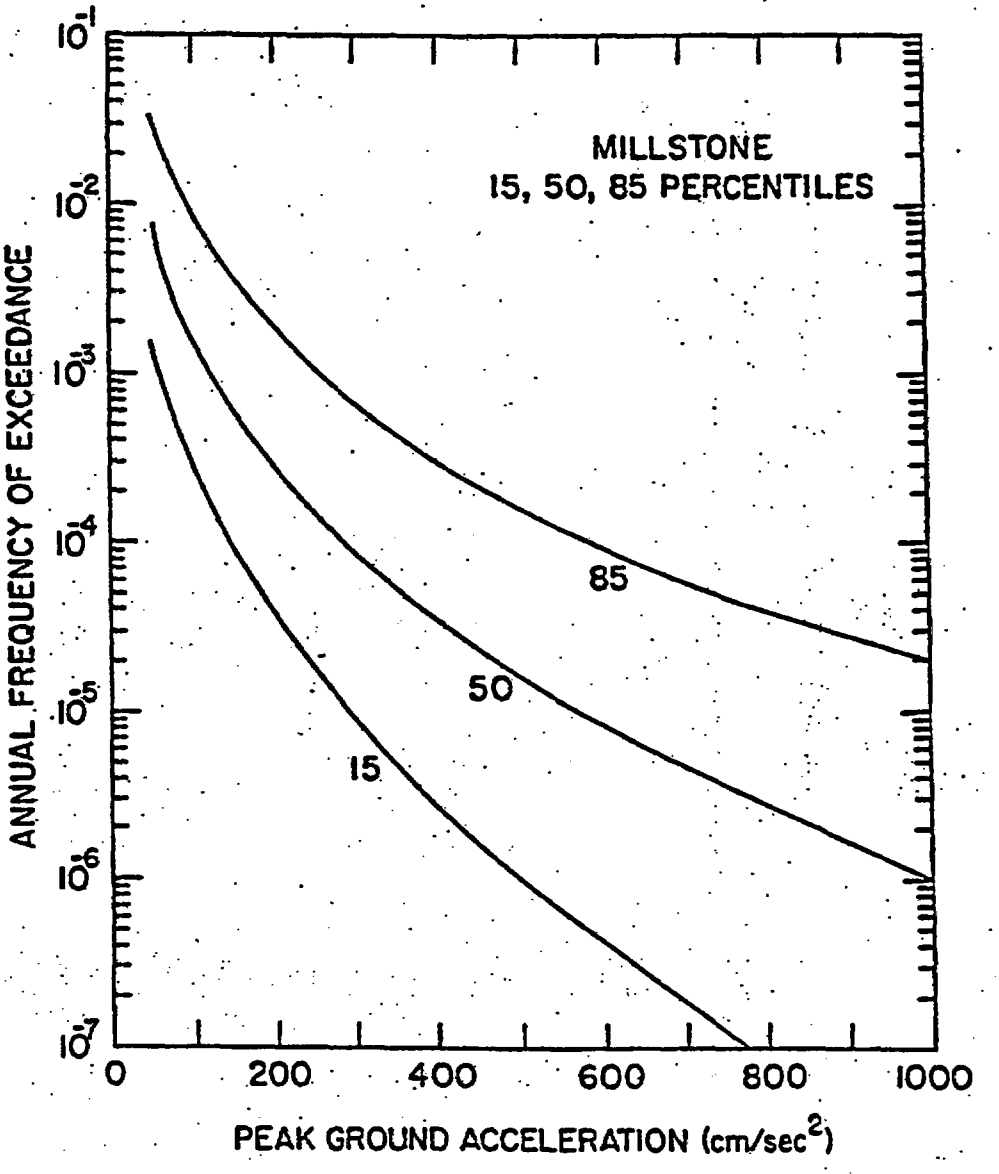

Figure 2.2 The 15, 50 and 85 Percentile Hazard Curves for the Millstone Site. The data are based on confidence levels in the input seismicity data of the experts and uncertainties in the best choice of ground motion models (Source: Ref. 11, pg. 5-45).

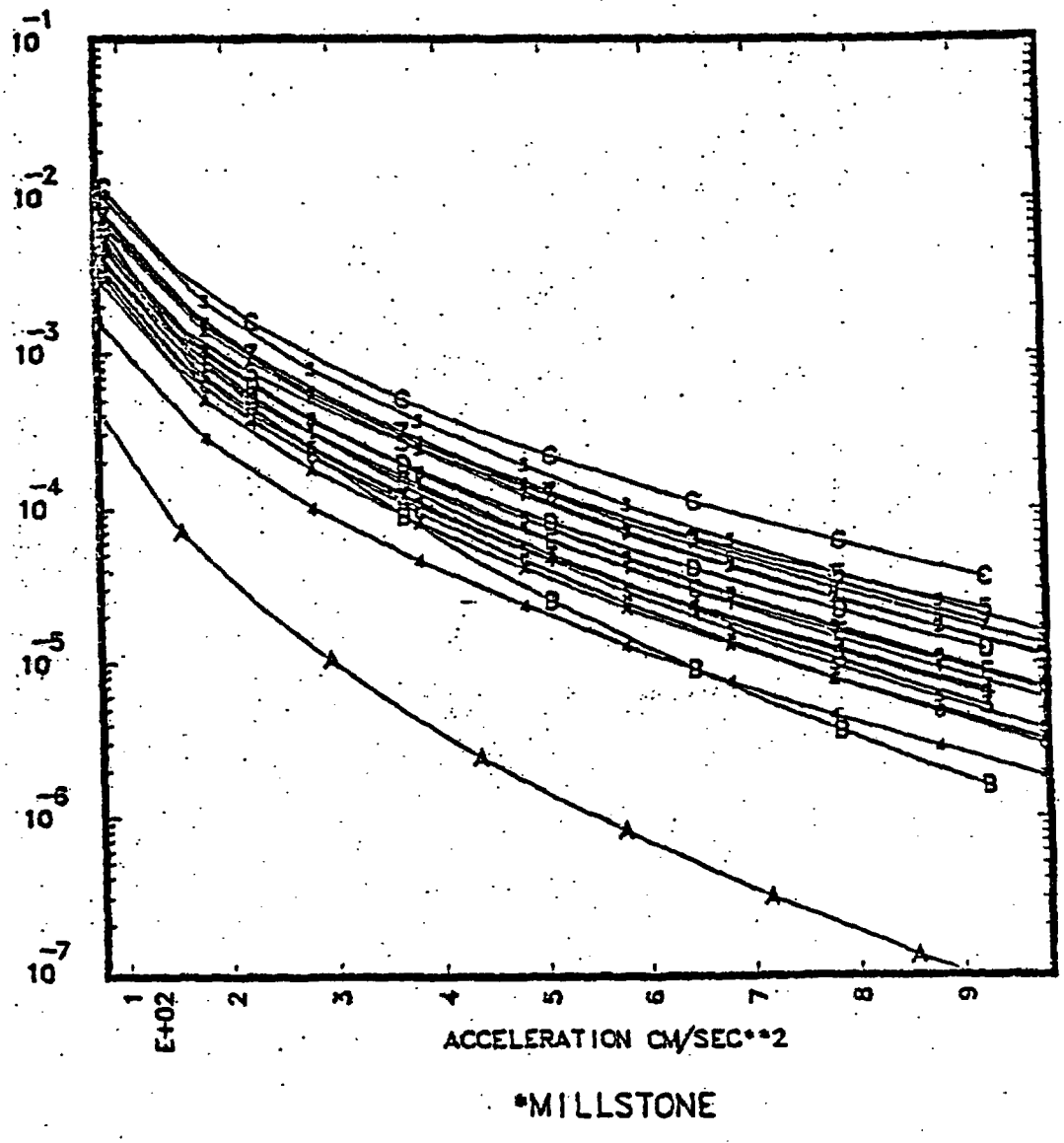

Figure 2.3 Seismic Hazard Curves for Millstone of Each of the Individual Experts Participating in the SEP Studies (Ref. 8) and/or the SHC Studies (Ref. 10). The curves give an indication of the spread in expert opinion (Source: Ref. 11, pg. 209).

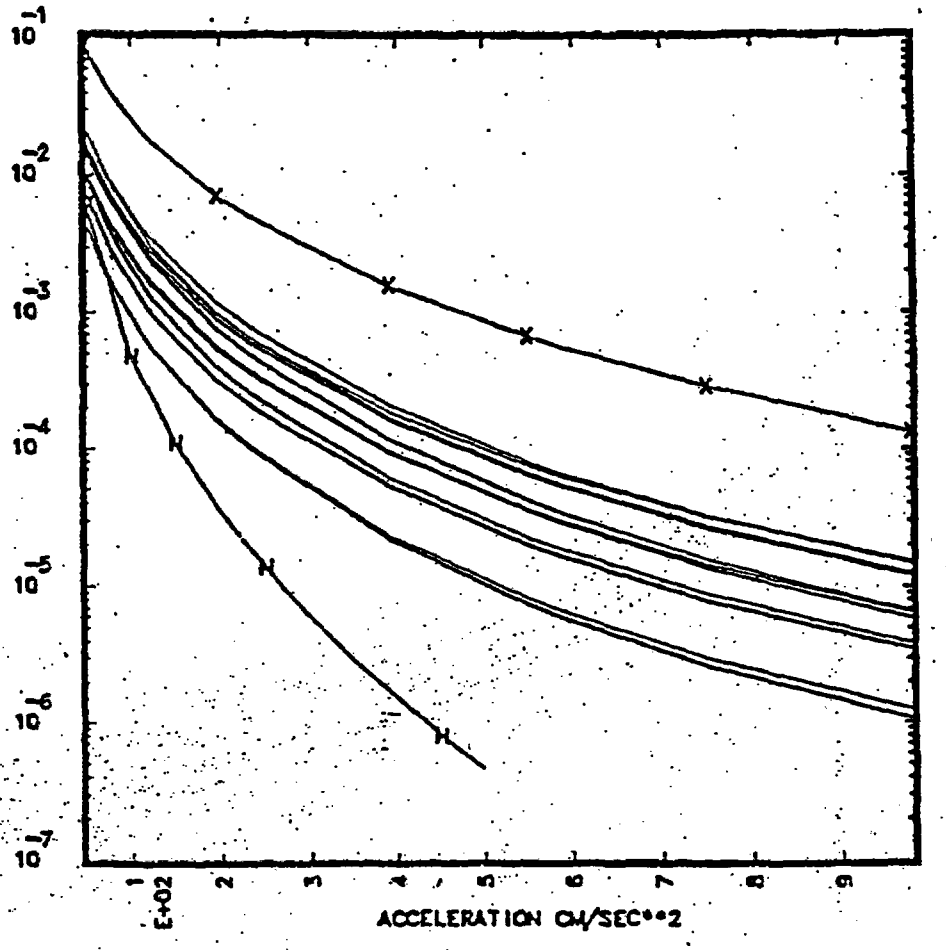

Figure 2.4 Comparison of the Millstone Site Hazard Curves Generated from the Data Input of the SHC Experts, with Those Generated from the USGS Data (Curve "X") and from the Historical Record of the Past 280 Years (Curve H) (Source: Ref. 11, pg. 6-7).

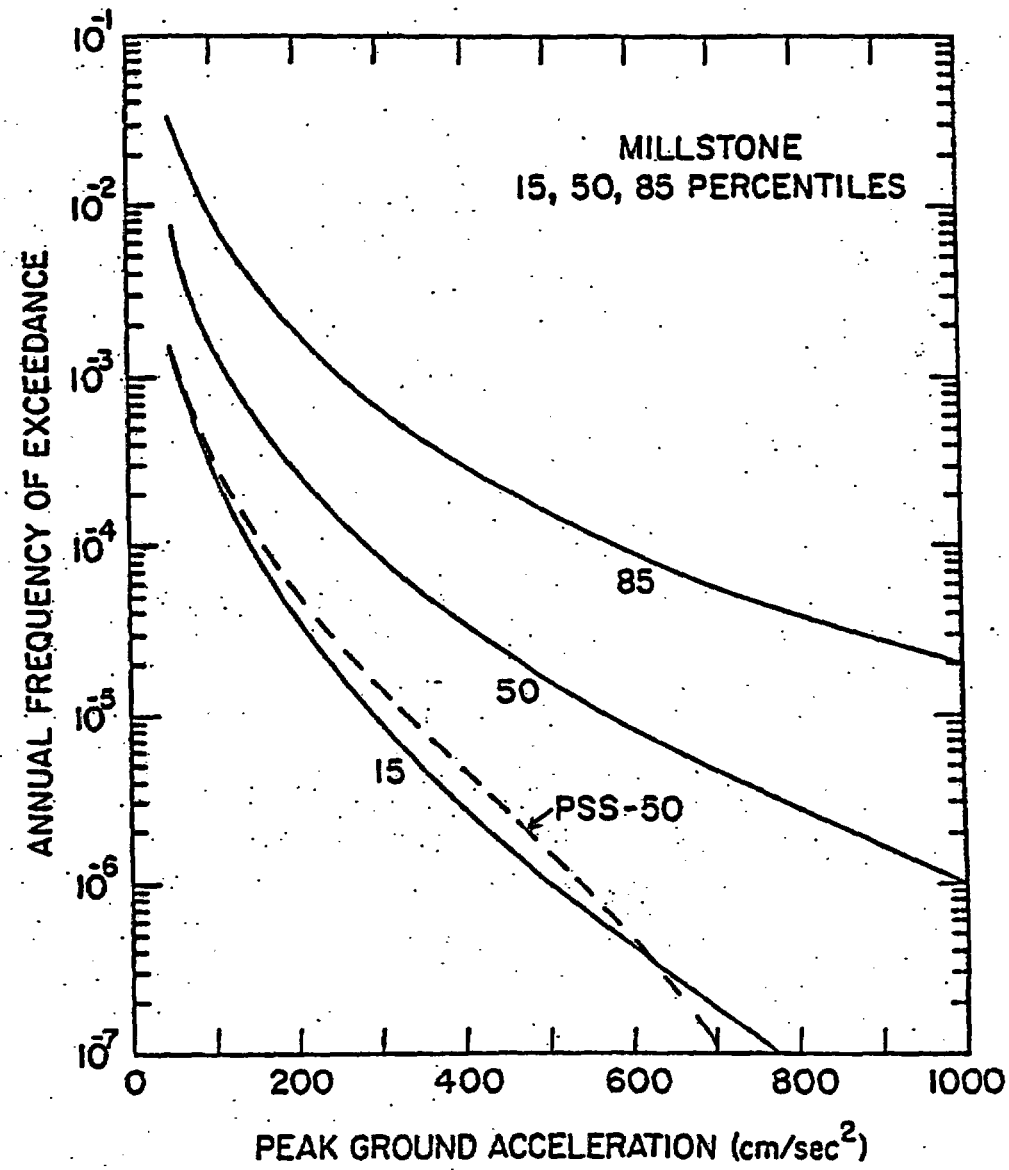

Figure 2.5 SHCP Seismic Hazard Curves of Fig. 2.2 Compared with the 50 Fractile Curve (PSS 50). Derived for the Millstone 3 Probabilistic Safety Study (Ref. 20).

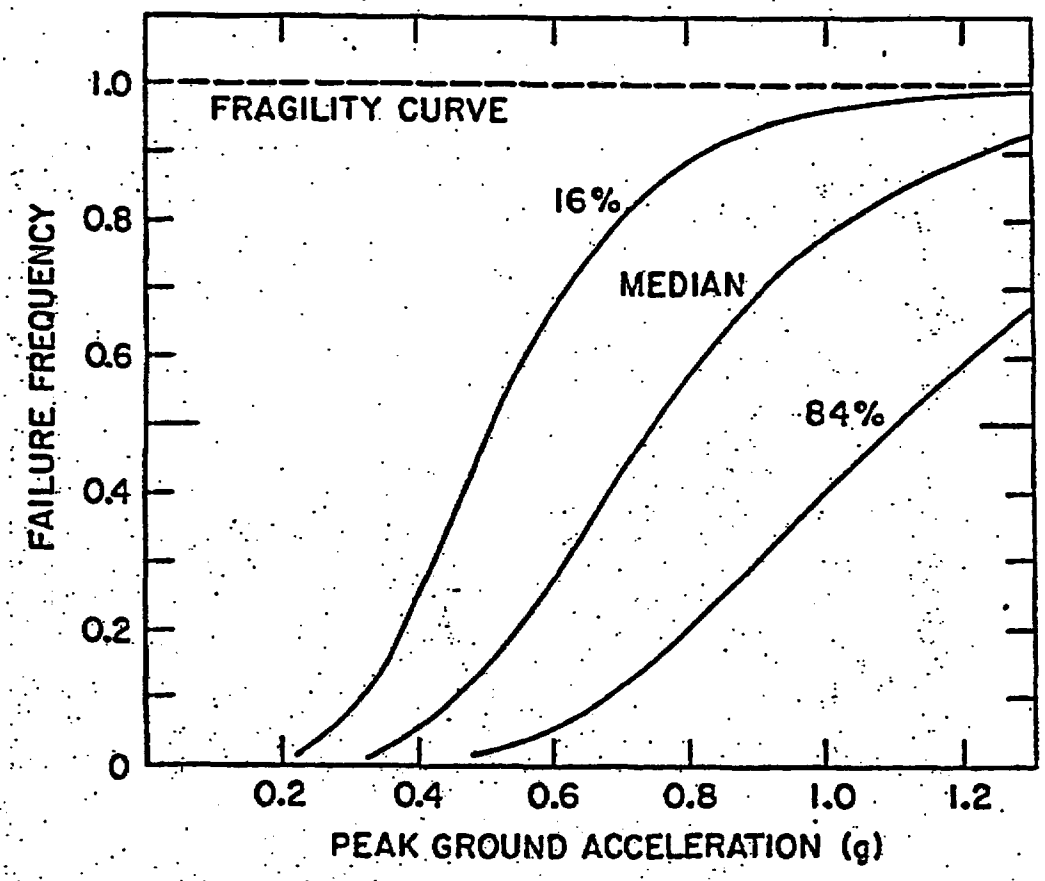

Figure 2.6 Fragility Curves for the Oyster Creek Reactor Building. The curves generated by R.P. Kennedy et al. (Ref. 23) give the frequency of structural failure as a function of peak ground acceleration during an earthquake.

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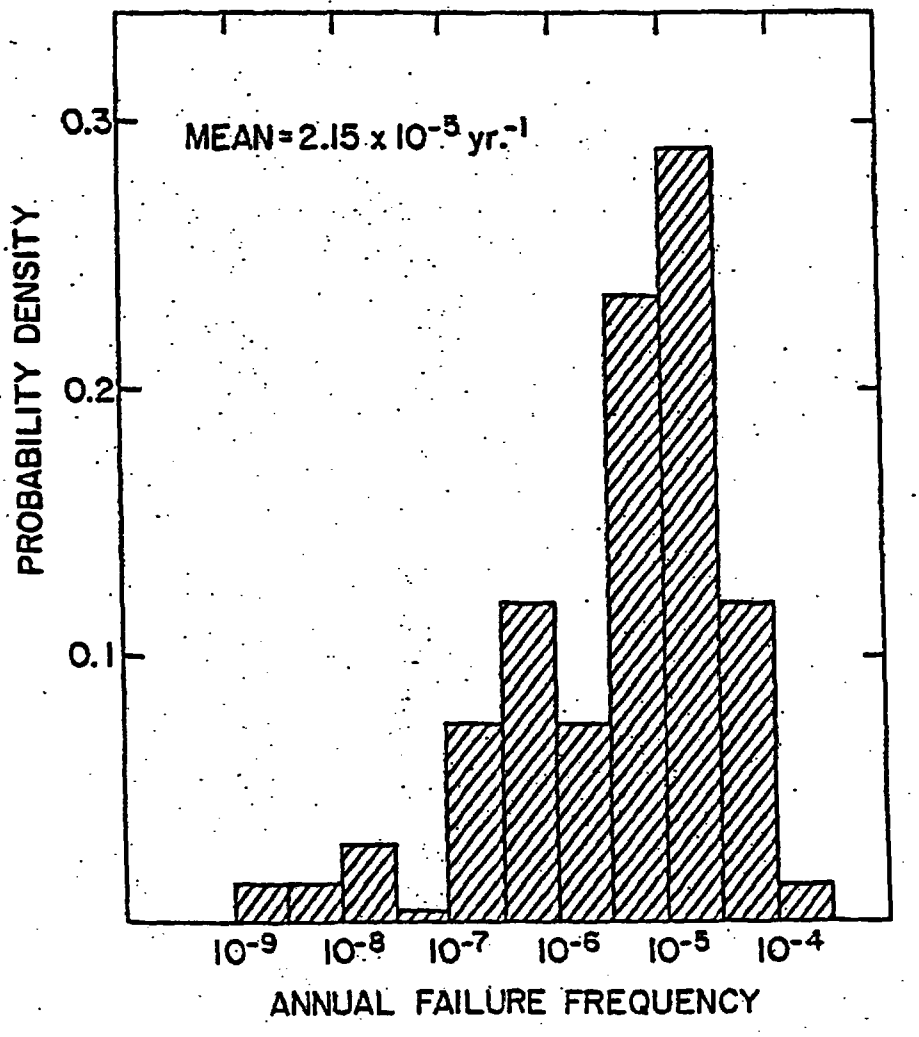

Figure 2.7 Probability Density for Seismically-Induced Failure as a Function of Annual Failure Frequency. The histogram was obtained from 24 convolutions of four hazard curves with six fragility curves and includes the weighting factors assigned to each curve.

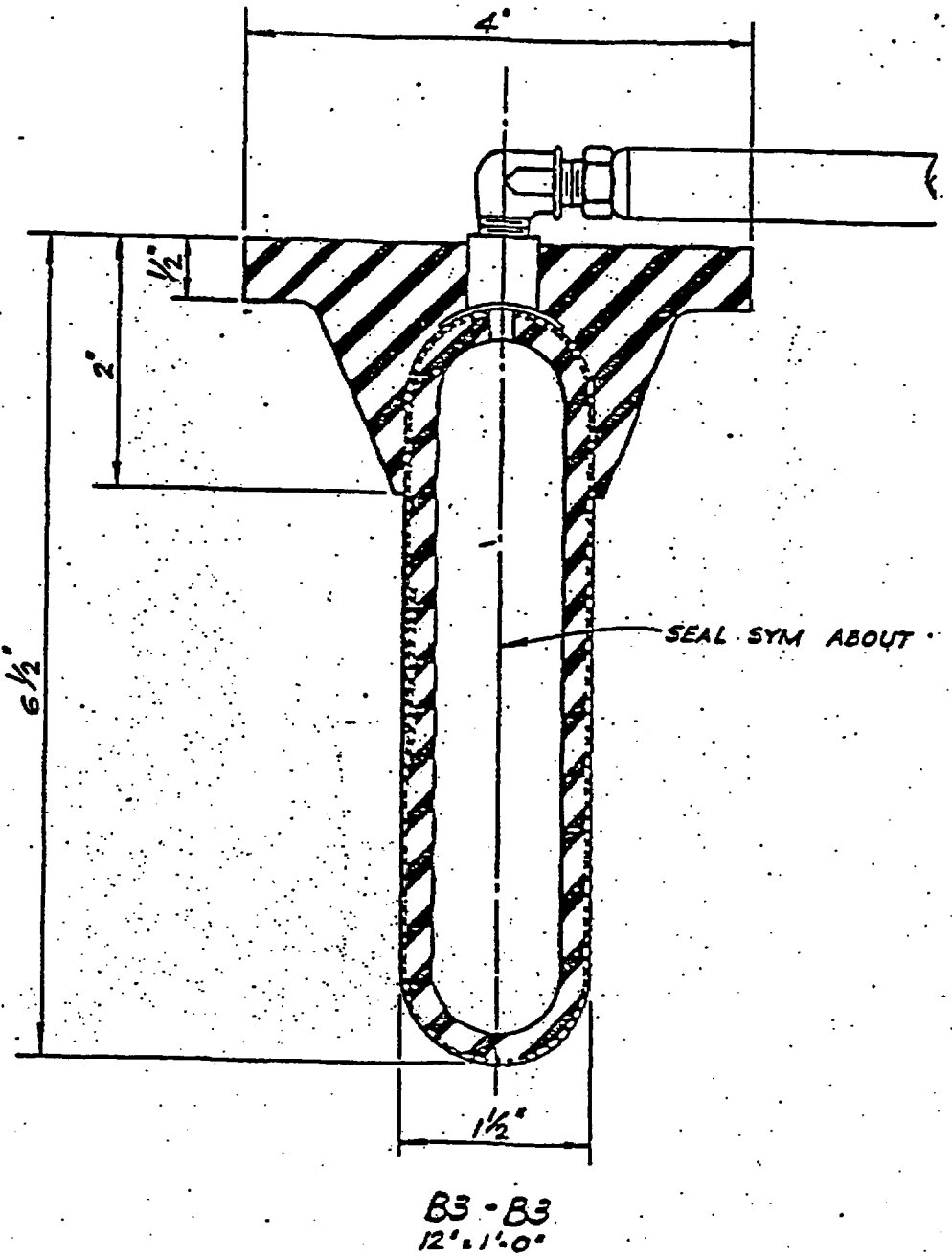

Figure 2.8 Cross Section of a Typical Pneumatic Seal (Source: submission by Sequoyah Nuclear Plant, Docket No. 50-327, 10/26/84 in response to IE Bulletin 84-03.

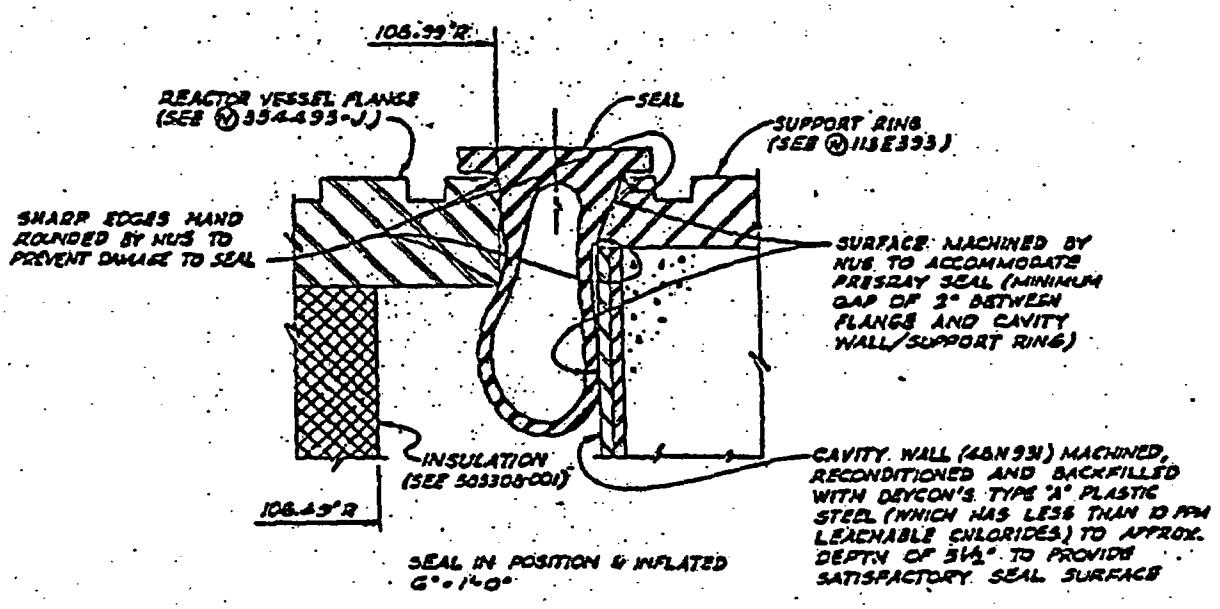

Figure 2.9 Cross Section of Inflated Pneumatic Seal Seated on the Reactor Vessel Flange and Inner Surface of Cavity Wall (Source: submission by Sequoyah Nuclear Plant, Docket No. 50-327, 10/26/84 in response to IE Bulletin 84-03.

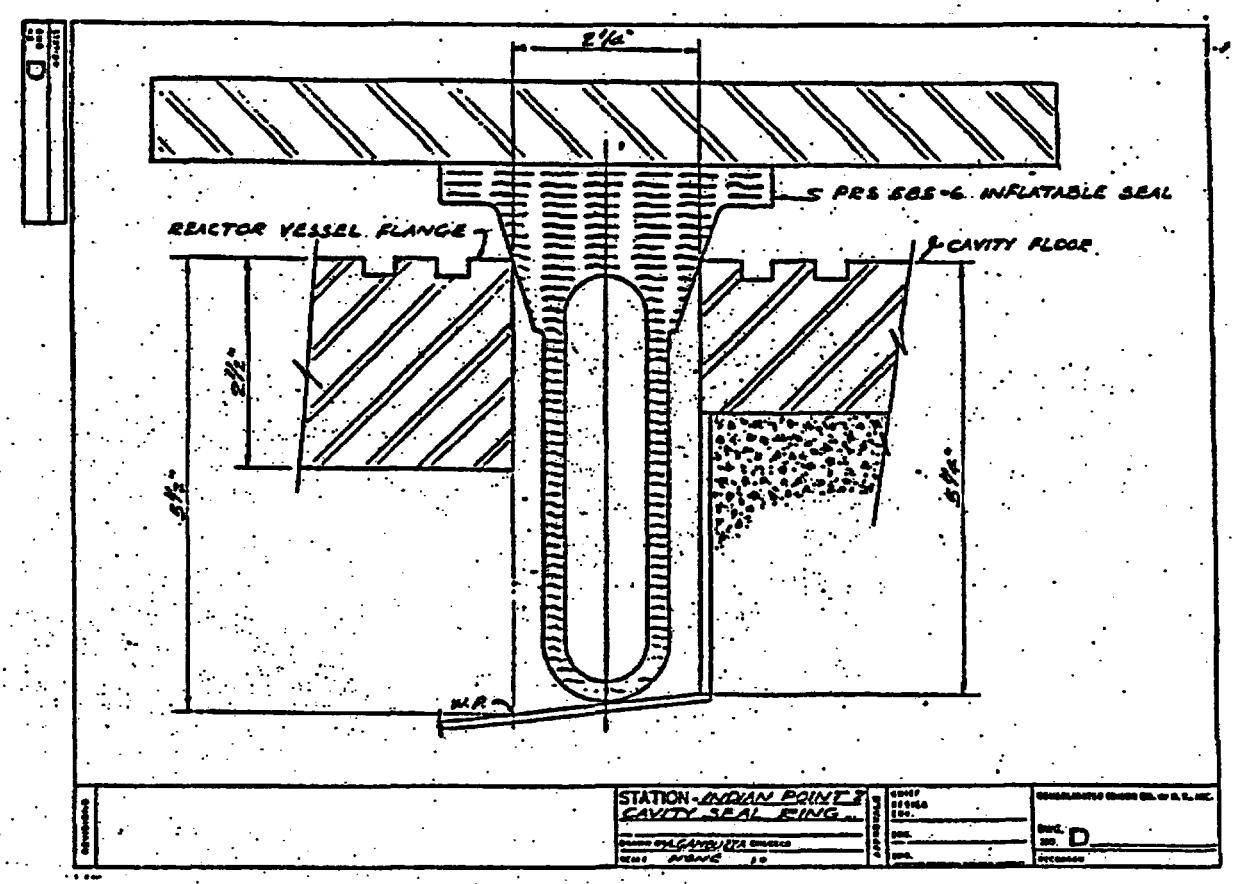

Figure 2.10 Uninflated Pneumatic Seal with Steel Hold-down Ring (Source: submission by Indian Point Station, Unit 2, Docket No. 50-247, 11/30/84 in response to IE Bulletin 84-03).

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3. EVALUATION OF FUEL CLADDING FAILURE

Two previous studies[1,2] employing the computer codes SFUEL and SFUEL1W, analyzed the thermal-hydraulic phenomena assuming a complete drainage of the water from a spent fuel pool. The previous section addressed the possible mechanisms for such an accident to occur and provided estimates for the accident frequency. This section provides a reevaluation of the results[1,2] obtained by SFUEL and SFUEL1W calculations and their applicability to beyond design-basis accidents in spent fuel pools.

3.1 Summary of SFUEL Results

The SFUEL code was developed by Benjamin et al.[1] to analyze the behavior of spent fuel assemblies after an accident has drained the pool. The results reported in Reference 1 indicated a wide range of decay power levels for which self-sustaining oxidation of the cladding would be predicted. Several limitations in the SFUEL model were identified and addressed in a subsequent investigation.[2] But comparisons to small scale experiments were not very successful.

Subsequent reviews of Zirconium/Zircaloy reaction rate data indicate that the oxidation equation used in SFUEL is representative of the existing data (see Appendix B). In this analysis it will be shown that self-sustaining oxidation initiation is not very sensitive to the oxidation rate equation but is dependent upon the calculated air flow (related to flow resistance) and the power level. BWR spent fuel with its low power density and open flow configuration must be recently discharged (within about 3 months) for self-sustaining oxidation to be initiated and unless it is a very high power bundle (discharged within 10 days or less) there is only a slight chance of propagation to older low power fuel bundles.

However, PWR spent fuel racks typically have a higher power density in storage and more flow restriction, thus self-sustaining oxidation may be initiated in fuel that has been discharged for one year or more.

3.1.1 Model Description

The SFUEL code was developed at SNL and is described in Reference 1. Basically it is a finite difference solution of the transient conduction equation for heating of the fuel rods considering:

- The heat generation rate from decay heat and oxidation of the cladding.
- Radiation to adjacent assemblies or walls.
- Convection to buoyancy-driven air flows.

The key assumptions in the analysis are:

- The water drains instantaneously from the pool.
- The geometry of the fuel assemblies and racks remains undistorted.
- Temperature variations across the fuel rods are neglected.
- The air flow patterns are one-dimensional.
- The spaces between adjacent basket walls are assumed to be closed to air flow.

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These assumptions simplify the analysis and may affect the timing and extent of fuel rod failure but they appear to provide a reasonable basis for the present "scoping" calculations. In particular, fuel pool failure, if it occurs, would not cause an instantaneous loss of water. Because of the large water inventory, draining of the pool would occur over several hours or days. The time to drain the pool would have a considerable impact on the likelihood of mitigative action and the timing of the fuel rod heat-up if mitigation were unsuccessful. The idealizations of undistorted geometry and one dimensional flow appear to be reasonable up until the point of clad melting and relocation. Degraded fuel rod phenomenology beyond the melting clad point is beyond the scope of this investigation but may also provide a mechanism for failure propagation to low power assemblies.

The last assumption of no air flow between baskets is accurate for high density configurations but may tend to underestimate cooling flow for some of the older designs (e.g., cylindrical baskets) with large air spaces between each basket.

After the water is drained from the pool the fuel rods heat up until the buoyancy driven air flow is sufficient to prevent further heating. If the decay heat level is sufficient to heat the rods to about 900°C, (1650°F) the oxidation becomes self-sustaining. That is, the exothermic oxidation reaction provides sufficient energy to match the decay heat contribution and the temperature rises rapidly.

Reference 2 describes the modification of the SFUEL code to increase calculational stability and assess propagation of Zircaloy "fires" from high power to low power assemblies. This version of the code (SFUEL1W) also eliminated unrealistically high temperatures* by non-mechanistically removing each node as it reaches the melting point of Zircaloy dioxide (2740°C or 4963°F). In the present investigation, the oxidation cutoff has been reduced to 1900°C (3450°F), which is the melting point of Zircaloy. Recent experiments indicate that Zircaloy relocation will restrict further oxidation occurring above the clad melting point.

3.1.2 Clad Fire Initiation Results

An extensive review of the cladding oxidation models used in SFUEL,[1,2] is given in Appendix B and summarized here:

1. The likelihood of clad fire initiation is not very sensitive to the oxidation equation.

2. The oxidation equation used in SFUEL is a reasonable representation of the data.

3. The likelihood of clad fire initiation is most sensitive to the decay heat level and the storage rack configuration (which controls the extent of natural convection cooling).

*Since the code does not explicitly treat melting of the cladding, temperatures as high as 3500°C were predicted.[2]

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The critical conditions (i.e., the decay heat level which is sufficient to cause a clad fire) for clad fire initiation as determined by SFUEL calculations are summarized in Table 3.1. Note that for the old style cylindrical fuel racks with a large inlet orifice (3 inch diameter) the natural convection cooling in air is predicted to be adequate to prevent self-sustaining oxidation (cladding "fires") after 10 days of decay for BWR assemblies and 50 days for PWR assemblies. However for the new high density fuel racks, natural convective flows are so restricted that even after cooling for a year there is potential for self-sustaining oxidation. As pointed out by Benjamin et al.[1] there are a number of modifications to the fuel rack design which would enhance convective cooling and reduce the potential for cladding fires. However, the limited flow area of the high density designs make it difficult to ensure adequate cooling by natural convection of air.

For the assumption of annual discharges, the critical decay time (the shutdown time which is sufficient to preclude clad fire) can be expressed as a fraction of the fuel cycle and translated into a likelihood of cladding fire for a complete loss of pool water inventory. For the critical cooling times given in Table 3.1 the fraction of time for which the decay heat is sufficiently high to cause self-sustaining oxidation is approximately:

0.0 to 0.5 for BWRs with low density storage racks,

0.0 to 0.7 for PWRs with low density storage racks, and

1.0 for PWRs with high density storage racks.

3.1.3 Clad Fire Propagation

The SNL investigations[1,2] of spent fuel behavior after a loss of pool integrity accident (assumed to result in complete drainage of the pool), identified a range of power levels necessary for the initiation of self-sustaining clad oxidation and substantially lower power levels at which adjacent fuel bundles would oxidize once oxidation had been initiated. However, the phenomenology of propagation is not well understood and there was considerable uncertainty in these estimates. Benjamin et al.,[1] Pisano et al.,[2] Han[3] and Johnsen[4] have pointed out a number of limitations in the previous analyses.[1,2] To put the present results in perspective it is worth mentioning the most important limitations, and what has been done in this study to reduce uncertainties.

1. The oxidation equation allows oxidation to continue beyond 1900°C (3450°F) where clad melting and relocation is expected. PBF and KfK tests show clad relocation at temperatures in the range of 1900° to 2200°C but the analyses have calculated temperatures as high as 3500°C (6330°F) without accounting for clad and fuel melting. At such high temperatures the radiation heat flux becomes very large and it is believed that the potential for propagation to adjacent bundles will be overestimated.

   To provide more realistic estimates of the potential for oxidation propagation, BNL has chosen to terminate oxidation at the Zircaloy melting point since recent severe accident research indicates that Zircaloy relocation will occur and further oxidation will be limited.

2. The SFUEL code had not yet been validated successfully against fuel rod oxidation data. A preliminary comparison[2] against SNL data was only partially successful.

   The SFUEL code has been compared to the SNL data in a separate section (3.2) and key portions of the code have been validated. Specifically, the axial heat up (without oxidation) and the temperature at which self sustaining oxidation is reached has been validated. If a low power spent fuel bundle heats up to within one or two hundred °C of self-sustaining oxidation due to its own internal energy there is a high likelihood that the additional energy from an adjacent high power bundle will be sufficient to bring it to the initiation point.

3. The reaction rate equation has been criticized as being too low for long term exposure at low temperatures (when oxide layers may flake off and expose fresh Zircaloy).

   Appendix B has shown that the SFUEL calculations are not very sensitive to the low temperature oxidation rate.

4. The lack of a fuel and clad melting and relocation model has also been criticized.

   The modified SFUEL code (SFUEL1W[2]) has sufficient flexibility to estimate the importance of oxidation propagation via radiation heat transfer to adjacent fuel racks. The subsequent behavior (e.g., fuel rod slumping) after the clad melting temperature is reached may provide an additional mechanism for propagation (heating from below), but this has been treated as an uncertainty in the extent of pool involvement.

5. Johnsen[4] criticized the clad failure criterion used in the SNL analyses.[1,2] He noted that the clad structural failure could occur at temperatures as low as 650°C if the thermal loading is sustained for several hours.

   In view of the large uncertainty in the thermal behavior, we agree that a prediction of temperatures in excess of 650°C should not be viewed as successful cooling of the assembly. At these temperatures cladding failure and fission product release is very likely and the potential for cladding "fires" is high due to the effects of asymmetric heating (from adjacent high power bundles). Two cases for which cladding failure is assumed without self-sustaining oxidation are presented in Chapter 4.

It should be emphasized that SFUEL does not address the question of Zircaloy oxidation propagation after clad melting and relocation. For recently discharged fuel (less than 90 days), or for severely restricted air flow (e.g., high density PWR spent fuel racks) the oxidation reaction is predicted to be very vigorous and failure of both the fuel rods and the fuel rod racks is expected. Thus a large fraction of the fuel rods would be expected to fall to the bottom of the pool forming a large debris bed. If water is not present in the bottom of the pool, the debris bed will remain hot and will tend to heat the adjacent assemblies from below. The investigation of debris bed

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formation is beyond the scope of the present study, but it appears to be an additional mechanism for oxidation propagation.

Propagation of cladding "fires" by particulate (i.e., spallation) or zirconium vapor transport has been investigated and eliminated in an approximate separate effects study by Pisano et al.[2] However, the SNL results using the modified SFUEL1W code indicate that propagation due to the heat flux (radiation and convection) from adjacent bundles is likely to occur even to very low power assemblies (at power levels corresponding to 3 years of decays).

With these considerations in mind, a series of SFUEL calculations were performed to establish the range of conditions for which propagation is predicted to occur. Both the power of the initiating bundle and the power of the adjacent bundles have been varied as well as the ventilating conditions of the spent fuel building. An example SFUEL input deck is provided in Appendix C for documentation purposes.

Two fuel building ventilation conditions have been investigated as described below but it must be recognized that both of these assumptions correspond to very idealized conditions that are unlikely to be duplicated in an actual accident. Rather these idealized conditions are provided to demonstrate the sensitivity of the various assumptions. For a beyond design basis seismic event, that ruptures the pool, it seems likely the failure of the fuel building may also occur. Benjamin et al.[1] have shown that a very large hole (at least 77 ft^2) must be opened to approximate the perfect ventilation case.

3.1.3.1 Perfect Ventilation

Under the perfect ventilation condition it is assumed that the fuel building is maintained at ambient conditions by a high powered ventilation system (note that the flow rate must be much higher than typical gas treatment systems) or by a large opening (greater than 77 ft^2) in the building. Oxygen is not depleted and the air entering the pool is assumed not to be heated by the hot gases exiting the fuel assemblies. The conditions necessary to initiate self-sustaining oxidation under perfect ventilation conditions are summarized in Table 3.1 for three typical fuel rack configurations. Note that these are "borderline" conditions in that a slightly lower power level or a larger inlet hole size would predict that self-sustaining oxidation does not occur. Note that the "critical" conditions outlined in Table 3.1 do not imply that fuel rod failure would not occur for power levels below these conditions. The power level must be reduced substantially (about 20%) to ensure that the predicted clad temperature is below 650°C (the minimum temperature at which clad structural failure and fission product release is likely to occur).

For power and flow conditions that are only slightly below the "critical" conditions it should be obvious that the heat flux from a much higher power adjacent bundle would have the potential to push the "non-critical" fuel over the self-sustaining oxidation threshold. Thus the only real propagation question is whether recently discharged (high power) spent fuel will radiate sufficient energy to initiate self-sustaining oxidation in low power fuel bundles that have been cooled for one or more years.

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In this context two limitations of the SFUEL1W[2] code should be noted:

1. The fuel storage racks are assumed to be immediately adjacent so that no air flow between racks is allowed. (The numerical approach used to calculate the heat transfer is numerically unstable if flow is allowed).

2. All fuel storage racks are assumed to be identical so that the question of propagation from high power cylindrical racks to low power high density racks cannot be addressed.

The first limitation probably represents current storage practices where a number of fuel pools are approaching their design capacity. However, the question of providing deliberate cooling channels between recently discharged fuel and the older fuel cannot be directly addressed. Based on engineering insight, it appears that, under the idealized perfect ventilation conditions, the provision of an air space of 6 to 12 inches around the periphery of recently discharged fuel would minimize the likelihood of oxidation propagation to low power spent fuel assemblies. (Note that the code does allow for an air space adjacent to the pool walls and 6 to 12 inches is found to be adequate if flow through the bundles is not restricted.)

Since high density fuel storage racks are predicted to cause self-sustaining oxidation even after storage for one or more years, it seems clear that it would be undesirable to store spent fuel in high density storage racks if it has been discharged within the last two years. (It may be worth noting that current practice restricts the storage density of low burnup fuel due to nuclear criticality considerations.) Thus the question of propagation from cylindrical fuel racks to high density fuel racks should be addressed, but the second limitation mentioned above precludes intermixing of the storage rack configurations.

The propagation results with perfect ventilation are summarized in Table 3.2 for the high density rack configuration described in Reference 2. Note that the lowest power (11.0 kW/MTU) for self-sustaining clad oxidation corresponds approximately to fuel that has been discharged for one year, but the oxidation reaction will generate sufficient energy to propagate to a fuel bundle that is about 2 years old (6.0 kW/MTU). For a fuel assembly that has been discharged for about 10 days (90 kW/MTU) the high decay heat level causes extensive clad oxidation in the high power bundle and a somewhat higher propensity to propagate to low power fuel assemblies (as low as 5 kW/MTU which corresponds roughly to a 2-1/2 year old discharge).

The propagation results for a low density fuel rack (cylindrical) with a 3 inch diameter inlet hole is summarized in Table 3.3. Note that the range of power for the high power assembly is limited due to the improved free convection within this type of fuel rack. Thus self-sustaining clad oxidation is initiated at decay power levels at or above 30 kW/MTU (corresponding to about 90 days of cooling). Assuming that more than one discharge per year is unlikely, the adjacent low power assembly must be less than or equal to about 19 kW/MTU (180 days of cooling). Thus propagation only occurs for fuel that has been discharged less than 1 year with initiation from fuel that has been discharged within 2 weeks.

For a PWR cylindrical fuel rack with only a 1.5 inch diameter flow hole, the air flow is much more restricted and the possibility of propagation is stronger as indicated in Table 3.4. For the 1.5 inch hole size propagation is predicted to occur for cooling times as long as two years.

3.1.3.2 Inadequate Ventilation

As previously mentioned the case of perfect ventilation implies a very high ventilation rate that is not normally possible. Benjamin et al.[1] extended the SFUEL code to consider limited heat removal to just keep the spent fuel building at constant pressure. Details of the modeling are described in Reference 1, but the main result of the model is that the fuel building atmosphere heats up (due to decay heat and the chemical energy of oxidation) and the oxygen is depleted. Benjamin[1] found that the heat-up of the building increased the likelihood of self-sustaining oxidation (i.e., decreased the decay power level necessary to initiate self-sustaining oxidation). This section is intended to address the question of whether limited ventilation also increases the likelihood of propagation to low power bundles.

Table 3.5 provides a summary of propagation runs under inadequate ventilation conditions. For the analyses the high power assemblies are modeled to represent approximately 1/3 of the core for 1000 MWe plant and the fuel building is taken to have a volume of 150,000 ft^3. The results given in Table 3.5 indicate that propagation is no more likely with inadequate ventilation than with perfect ventilation. In fact propagation does not occur for several conditions listed in Table 3.5 for which propagation was predicted with perfect ventilation. Although this result is somewhat surprising, it is simply a result of the oxygen depletion calculation. That is, the oxidation of the recently discharged assemblies uses up the oxygen supply before the lower power assemblies can be heated to the point of self-sustaining oxidation.

In view of the potential for fuel building failure due to either the assumed initiating event (e.g., a beyond design basis earthquake) or the rapid building pressurization from Zircaloy combustion and decay heat, BNL considers the oxygen depletion calculation to be unrealistic. Thus, in spite of the many uncertainties, the perfect ventilation model is expected to give the best approximation for the potential for propagation.

Conclusions Regarding Propagation

Based on the previous results we have concluded that the modified SFUEL code (SFUEL1W[2]) gives a reasonable estimate of the potential for propagation of self-sustaining clad oxidation from high power spent fuel to low power spent fuel. Under some conditions, propagation is predicted to occur for spent fuel that has been stored as long as 2 years.

The investigation of the effect of insufficient ventilation in the fuel building indicated that oxygen depletion is a competing factor with heating of the building atmosphere and propagation is not predicted to occur for spent fuel that has been cooled for more than three years even without ventilation.

These results are in general agreement with the earlier SNL studies,[1,2] but tend to show a reduced likelihood of clad fire propagation due to the reduced oxidation cutoff temperature (2100°C) used in the BNL analysis.

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3.2 Validation of the SFUEL Computer Code

The SNL investigations[1,2] of spent fuel behavior after a loss of pool integrity accident (assumed to result in complete drainage of the pool), identified a range of power levels necessary for the initiation of self-sustaining clad oxidation and substantially lower power levels at which adjacent fuel bundles would oxidize once oxidation had been initiated. However, an attempt[2] to validate the code was only minimally successful in that the post-test analyses were able to match the heat-up rate in helium (without oxidation) but the SFUEL code over-estimated the temperature transient after air was introduced.

The objective of this section is to use the revised[3] oxidation rate equation in SFUEL to analyze the SNL small scale tests to aid in validating the SFUEL code. The SNL tests are described in Reference 2, but in order to put the test results in perspective several important conditions should be highlighted:

1. The test was of a small bundle of electrically heated rods (9 rods) with a short length (38 cm).

2. In order to achieve self-sustaining clad oxidation (>850°C) the rods were heated with a very low flow rate of helium before air was admitted to the test assembly.

Under these test conditions the dominant heat loss is via radiation whereas for the postulated accident the dominant heat loss is via free convection. These test conditions lead to laminar flow (a Reynolds number of about 100) in which oxygen diffusion to the cladding surface limits the reaction rate. Only one test (6) had a sufficiently high air flow rate to allow vigourous oxidation.

Since the free convection and radiation calculations in SFUEL[1,2] were inappropriate to the test configuration, Pisano et al.[2] created a stripped down version called CLAD[2] which used a matrix inversion routine to calculate radiation losses.

After several preliminary attempts to analyze the helium portion of the tests BNL concluded that there were several errors which led to underestimation of the convection portion of the heat losses. Since helium has a much higher heat capacity and conductivity than air it appears to contribute to establishing the initial conditions. In order to provide an adequate simulation of the initial steady-state portion of the test we made two modifications to the CLAD code:

1. Include helium properties with a switch to air properties at the start of the transient.

2. Include an energy balance on each gas control volume to force conservation of energy.

With these changes we were able to obtain an adequate simulation of the initial portion of the tests. Using this revised version of CLAD with the

Weeks' oxidation correlation,[3] analysis of both the helium and the air portions of the test looked reasonable, but still tended to over-predict the peak temperatures during oxidation. In order to bring the calculations into reasonable agreement with the small scale data the Weeks' correlation has been reduced by a factor of four (note that this corresponds approximately to the data scatter).

Results

The revised CLAD code has been used to analyze the SNL small scale experiments Tests 4, 5 and 6. The other three tests were intended to simulate propagation with nonuniform heating and structures that CLAD was not capable of modeling. The CLAD results for Test 4 are compared to the data in Figure 3.1. These results still tend to overpredict the temperature in the center of the test rod, but give reasonably good agreement at the top of the rod where radiative heat losses are large.

The peak temperatures calculated by CLAD are summarized in Table 3.6 and compared to the peak measured temperatures for the three tests. Note that CLAD still overpredicts the peak temperature for the low flow rate tests (4 and 5) but gives good agreement with the high flow rate tests where adequate oxygen is available. It should be noted that this "oxygen starvation" phenomenon appears to be a result of the extremely low laminar flow where oxygen must diffuse to the clad surface. CLAD includes an oxygen depletion calculation but assumes that all the oxygen in each volume is immediately available at the surface.

3.3 Conclusions Regarding SFUEL Analyses

After an extensive review of the SFUEL code and comparison to the SNL small scale experiments, BNL concludes that the code provides a valuable tool for assessing the likelihood of self-sustaining clad oxidation for a variety of spent fuel configurations assuming that the pool has been drained.

The SNL small scale data provide a reasonable degree of validation for the heat-up and oxidation models, but the results are extremely sensitive to the natural convection calculation which has not been validated.

When oxidation is terminated at the Zircaloy melting temperature (assuming that the molten Zircaloy is relocated), oxidation propagation only occurs for spent fuel bundles which are already approaching the "critical" conditions for self-sustaining oxidation (see Table 3.1). However, this finding does not mean that oxidation propagation is unlikely. On the contrary, for some high density storage configurations the "critical" conditions are approached for spent fuel that has decayed for two to three years. Thus clad "fire" propagation appears to be a real threat but the basic question remains as to what are the "critical" conditions for initiation of oxidation and what the uncertainty is for a given spent fuel configuration. The critical conditions are summarized in Table 3.1 for several typical spent fuel racks. While the heat-up and oxidation models have been validated to a limited extent by the SNL data (see Section 3.2), the authors believe that the largest source of uncertainty is in the natural convection flow rate. It is recommended that these free convection flow calculations be verified against large scale data. Preferably the

data would be obtained from spent fuel assemblies in typical storage racks (both high and low density).

3.4 References for Section 3

1. A.S. Benjamin, D.J. McCloskey, D.A. Powers, S.A. Dupree, "Spent Fuel Heat-up Following Loss of Water During Storage," NUREG/CR-0649, March 1979.

2. N.A. Pisano, F. Best, A.S. Benjamin, K.T. Stalker, "The Potential for Propagation of a Self-Sustaining Zirconium Oxidation Following Loss of Water in a Spent Fuel Storage Pool," Draft Report, January 1984.

3. J.T. Han, Memo to M. Silberberg, USNRC, May 21, 1984.

4. G.W. Johnsen, Letter to F.L. Sims, EG&G, Idaho, April 4, 1984.

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Table 3.1 Summary of Critical Conditions Necessary to Initiate Self-Sustaining Oxidation

Spent Fuel Rack Configuration	Inlet Orifice Diameter (inches)	Minimum Decay Power (kW/MTU)	Approx. Critical[1] Decay Time (days)
High Density PWR (6 assemblies per rack)	5	6	700
High Density PWR (6 assemblies per rack)	10	11	360
Cylindrical PWR	5	90	10
Cylindrical PWR	3	45	50[2]
Cylindrical PWR	1.5	15	250[2]
Cylindrical BWR	1.5	14	180
Cylindrical BWR	3.0	70	<10

[1] Critical cooling (decay) time is the shutdown time necessary to reach a decay power level below the minimum decay power for self-sustaining oxidation. The cooling time to prevent cladding failure is at least 20% longer.

[2] Note that these critical cooling times are somewhat lower than that found by Benjamin et al.[1] since the orifice loss coefficient was modified at BNL

Table 3.2 Summary of Radial Oxidation Propagation Results for a High Density PWR Spent Fuel Rack with a 10 Inch Diameter Inlet and Perfect Ventilation

High Power Level (kW/MTU)	Adjacent Power Level (kW/MTU)	Approximate Decay Time (days)	Propagation
11.0	5.9	365	Yes
19.2	5.9	365	Yes
90	5.9	365	Yes
90	4.0	730	No

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Table 3.3 Summary of Radial Oxidation Propagation Results for a Cylindrical PWR Spent Fuel Rack with a 3 Inch Diameter Hole and Perfect Ventilation

High Power Level (kW/MTU)	Adjacent Power Level (kW/MTU)	Approximate Decay Time (days)	Propagation
90	11.0	365	No
90	19	180	Yes*

*Note that this is an unlikely situation in that the conditions imply a six month period between discharges.

Table 3.4 Summary of Radial Oxidation Propagation Results for a Cylindrical PWR Spent Fuel Rack with a 1.5 Inch Diameter Hole and Perfect Ventilation

High Power Level (kW/MTU)	Adjacent Power Level (kW/MTU)	Approximate Decay Time (days)	Propagation
90	11.0	365	Yes
90	5.9	730	Yes
90	3.0	1100	No
15	11.0	365	Yes
15	5.9	730	No

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Table 3.5 Summary of Radial Oxidation Propagation Results for Various PWR Spent Fuel Racks with No Ventilation

Spent Fuel Rack	High Power Level (kW/MTU)	Adjacent Power Level (kW/MTU)	Propagation
Cylindrical with 1.5 inch hole	90	5.9	Yes
Cylindrical with 1.5 inch hole	90	3.0	No (O_2 depletion)
Cylindrical with 3 inch hole	90	5.9	No
Cylindrical with 3 inch hole	19.2	11.0	Yes
High Density with 10 inch hole	90	4.0	No (O_2 depletion)

Table 3.6 Comparison of SNL Small Scale Oxidation Tests to Calculations with CLAD

Test	Air Flow Rate (lpm)	Peak Temperatures Data (°C)	CLAD (°C) Mid	CLAD (°C) Top
4	12	1570	1900	1400
5	28.3	~1850	1960	1660
6	56.6	>2000*	2100	1800

*Thermocouple failure.

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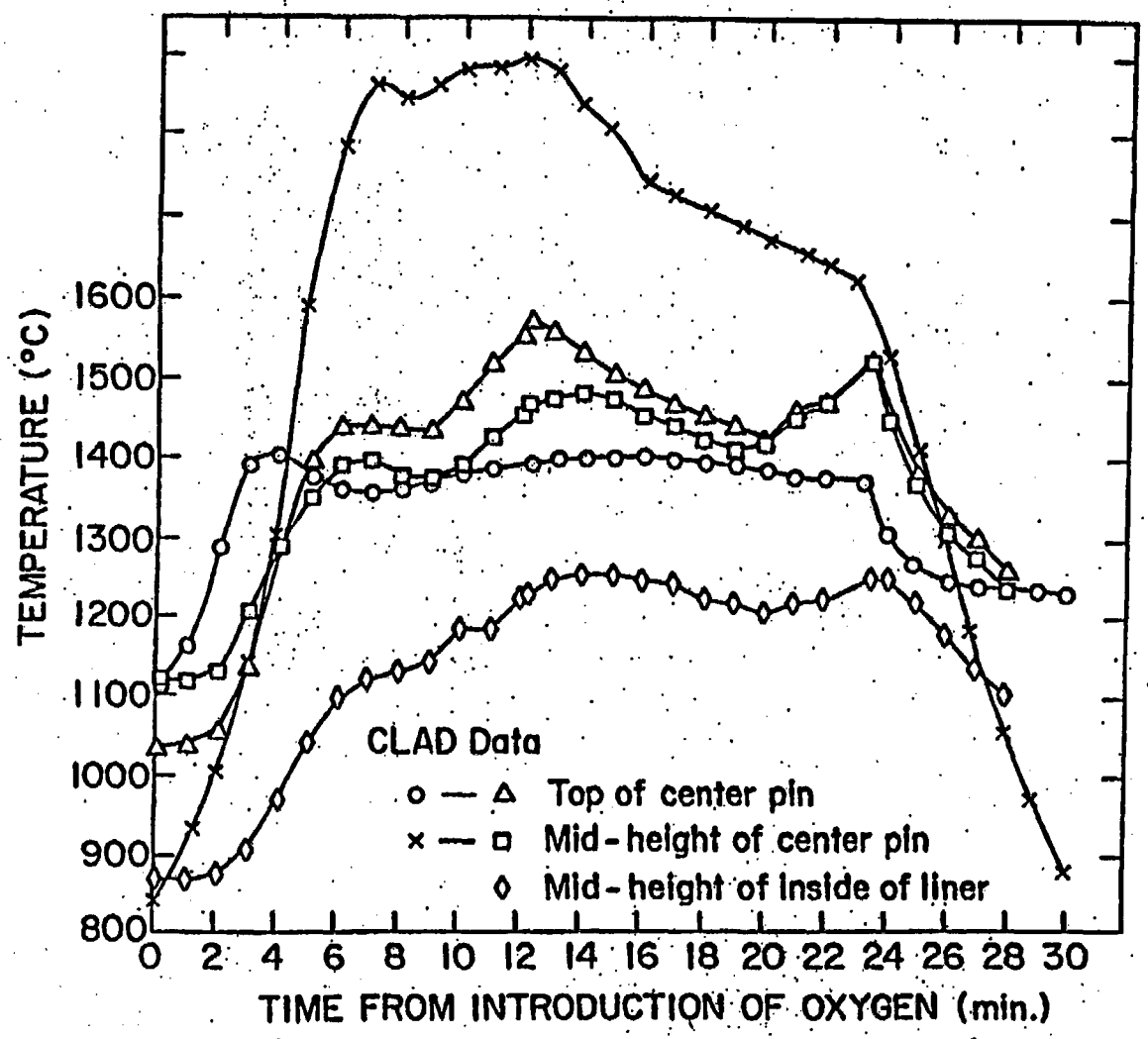

Figure 3.1 Comparison of CLAD to SNL data for Test 4.

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4. CONSEQUENCE EVALUATION

A PWR and a BWR reactor were selected for risk evaluation based on a preliminary screening[1] of perceived vulnerability and the spent fuel pool inventory. The reactors selected were Ginna and Millstone 1. Both are older plants that were built before the current seismic design criteria were promulgated and have relatively large inventories of spent fuel.

4.1 Radionuclide Inventories

The radionuclide inventories for both the PWR and BWR pools were calculated using the ORIGEN2 Computer Code[2] for the actual operating and discharge histories for Ginna and Millstone 1. The ORIGEN2 program in use at BNL was verified by comparison with results obtained at ORNL for identical cases.[3]

A description of the assumptions and methods of analysis is given in Appendix A along with the detailed results for each species. The results for the risk significant species are summarized in Table 4.1 (Millstone 1) and Table 4.6 (Ginna).

For both plants, the noble gases and halogens in the spent fuel inventories are a small fraction of the inventory in an equilibrium core at shutdown except for freshly discharged fuel, but cesium and strontium are more than three times the equilibrium inventory (see Tables 4.1 and 4.6).

4.2 Release Estimates

The fission product release fractions have been calculated for two limiting cases in which a Zircaloy fire occurs: In Case 1, the clad combustion is assumed to propagate throughout the pool and the entire inventory is involved. In Case 2 only the most recently discharged fuel undergoes clad combustion.

The release calculations for Cases 1 and 2 make the assumption that if the spent fuel pool suffers a structural failure, coolant inventory will be totally drained, i.e., the leak rate will greatly exceed makeup capability even if the coolant systems are still available. The probability of Zircaloy fire and fission product release has been determined from BNL calculations described in Section 3. In order for a cladding fire to occur the fuel must be recently discharged (about 10 to 180 days for a BWR and 30 to 250 days for a PWR). If a one year refueling cycle is assumed, the SFUEL results in Section 3 indicate that the fraction of time that the fuel must be cooled to preclude overheating leads to a mean conditional probability for a Zircaloy fire of about .25 for a BWR and .4 for a PWR. If the discharged fuel is put into high density racks the air cooling capability is limited such that the critical cooling time is increased to one to three years and the conditional probability of a Zircaloy fire is increased to nearly 1.0.

The BNL reevaluation of the cladding fire propagation calculations with SFUEL (see Section 3) indicates that there is a substantial likelihood of propagation to other fuel bundles that have been discharged within the last one or two years. Subsequent propagation to low power bundles by thermal radiation is highly unlikely, but with a substantial amount of fuel and

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cladding debris on the pool floor, the coolability of even low power bundles is uncertain.

4.2.1 Estimated Releases for Self-Sustaining Cladding Oxidation Cases (Cases 1 and 2)

As discussed in Section 3.1 there are a broad range of spent fuel storage conditions for which self-sustaining oxidation of the cladding will occur if the water in the pool is lost. For Ginna with high density racks the conditional probability of a cladding "fire" is predicted to be nearly 100% while for Millstone 1 the probability is about 25%. If self-sustaining oxidation occurs the fuel rods are predicted to reach 1500 to 2100°C over a substantial portion of their length. At these temperatures, the release fraction is predicted to be substantial.

Rough estimates of the fractional release of various isotopes have been presented in an attachment to Ref. 4. Included in the estimates were noble gases (100%), halogens (100%), alkali metals (100%), tellurium (2 to 100%), barium (0.2%), strontium (0.2%) and ruthenium (0.002%).

Estimated release fractions of other isotopes are given in Table 4.2. These estimates are based on empirical fission product release correlations used in the CORSOR code.[5] The uncertainty range included in Table 4.2 is based upon the CORSOR coefficient for the reduced fuel rod temperatures predicted by the modified SFUEL1W code. Comments on the estimates listed in Table 4.2 follow:

Cesium: The uncertainty range in the cesium releases include a decontamination factor of 2 to 10.

Tellurium: The releases shown assume the lower limit of Ref. 4 based on the tellurium release model recently proposed by Lorenz, et al.[6] The low release value assumes that a fraction of the Zircaloy cladding relocates (melts and flows downward) before oxidation is complete.[6]

Alkali Earths: Because of the high boiling points of the oxides of Sr and Ba, it is estimated that only a very small fraction (2×10^{-3}) of these elements of fission product origin in the fuel pellets escape. It is estimated that 100% of the activation product Sr-89 and Y-91 contained in the Zircaloy cladding are released as aerosols.

Transition Elements: It is estimated that 100% of the transition element activation products contained in the cladding are levitated as aerosols of the oxides (smoke). Note that the small release fraction of Zr-95 (0.01) takes into account the large inventory of fission product Zr-95 trapped in the fuel pellets.

It is assumed that only 10% of the activation products in the assembly hardware escapes (see Table 4.2, Fe-55, Co-58, Co-60 and Y-91). The Co-60 fraction is corrected for its small content in the cladding.

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Antimony: It is estimated that 100% of the Sb-125 is roasted out of the fuel pellets, because of its high mobility.

Lanthanides and Actinides: A negligible release of the oxides of the lanthanides and actinides is estimated because of their chemical stability, low vapor pressures and ceramic characteristics.

Case 1: Case 1, the "worst" case, assumes an accident that results in a Zircaloy fire that propagates throughout the entire spent fuel inventory in the pool, and that the accident occurs 30 days after the reactor was shut down for discharge of the last fuel batch. The estimated releases of radionuclides are listed in Table 4.3. These were obtained by combining the "30-day" inventory given in Column 3 of Table 4.1 with the release fractions listed in Table 4.2.

Case 2: Case 2 assumes an accident that results in a Zircaloy fire that involves only the last fuel batch to be discharged, and that the accident occurs 90 days after the reactor was shut down for fuel discharge. The estimated releases of radionuclides are listed in Table 4.4. These were obtained by combining the inventory in the last fuel batch (data tabulated in Table A.6 of Appendix A) with the release fractions in Table 4.2.

4.2.2 Estimated Release for Low-Temperature Cladding Failure (Cases 3 and 4)

For a less severe accident in which fuel is exposed to air but does not reach temperatures at which a Zircaloy fire ignites, it is assumed that the cladding on many fuel rods will fail (i.e., develop leaks) resulting in a release limited to the noble gases and halogens. Two limiting cases have been considered:

Case 3: in which the entire pool is drained but the decay time since the last discharge is one year, and 50% of the fuel rods suffer clad rupture.

Case 4: in which the pool drains to a level that exposes the upper portion of the fuel assemblies, the decay time for the last discharged fuel batch is 30 days, no Zircaloy fire occurs but all of the fuel rods in the last discharged batch rupture.

The estimated releases for Cases 3 and 4 are given in Table 4.5.

4.3 Off-Site Radiological Consequences

4.3.1 Scenarios for Consequences Calculations

The off-site radiological consequences have been calculated using the CRAC2 computer code.[7] The scenario used in the CRAC2 calculations consisted of the following conditions:

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- a generalized site surrounded by a constant population density of 100 persons per square mile;
- generalized meteorology (a uniform wind rose, average weather conditions); and
- the population in affected zones was relocated after 24 hours.

The radiological effects were calculated out to distances of both 50 and 500 miles.

CRAC2 calculations were made for a range of possible releases as described in Section 4.2. While CRAC2 has been used extensively at BNL and elsewhere to calculate consequences for core melt accidents, its use for spent fuel accidents is subject to considerable interpretations. A sample input file is provided in Appendix C for the purpose of documentation of the present usage. The consequence calculations are summarized in Table 4.7.

4.3.2 Consequence Results

There are several unusual characteristics of a spent fuel accident that cause anomalous results with respect to the radiation dose calculations. Specifically, the calculated radiation exposure is insensitive to fairly large variations in the estimated release. This is due principally to the health physics assumptions within CRAC. For the long lived isotopes (predominantly cesium), the exposure is due mainly to exposure after the area is decontaminated and people return to their homes. The CRAC code assumes that decontamination will limit the exposure of each person to 25 rem. Thus, for this type of release the long term whole body dose is limited by the population in the affected sectors (about 0.8 million people in the 16 sectors for a 50 mile radius) to about 3×10^6 person-rem (only 3 of the 16 sectors are downwind). The extreme cases (1A; immediately after refueling and 1B and 1C; with the total fuel pool inventory involved) result in much higher releases but no significant change in population dose.

A more sensitive indication of the severity of a spent fuel accident is the interdiction area (the area with such a high level of radiation that it is assumed that it cannot ever be decontaminated). For these long lived isotopes the interdicted area increases directly with the release fraction and provides a convenient measure of the societal consequences. As indicated in Table 4.7 the worst spent fuel accident is calculated to result in an interdiction area of 224 sq. miles.

For the nominal cases (2A and 2B) in which propagation is assumed not to occur, the person-rem exposure is still high even with an assumed decontamination factor of 10 (Case 2B). The interdiction area is seen to be reduced substantially for Case 2B.

For cases in which the air cooling is sufficient to prevent clad fires (Case 3), the bulk of the releases are noble gases and the consequences are very small.

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4.4 References for Section 4

1. Internal Memorandum, Brookhaven National Laboratory, V.L. Sailor and K.R. Perkins to W.T. Pratt, "Study of Beyond Design Basis Accidents in Spent Fuel Pools," May 8 1985.

2. A.G. Croff, "ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials," Nuclear Technology, Vol. 62, pp. 335-352, September 1983.

3. Internal Memorandum, Brookhaven National Laboratory, from V.L. Sailor to R.A. Bari, "Comparison of BNL ORIGEN2 Calculations with ORNL," May 27, 1986.

4. Memorandum of J.T. Han to M. Silberberg, "Response to a NRR Request to Review SNL Studies Regarding Spent Fuel Heatup and Burning Following Loss of Water in Storage Pool," U.S. Nuclear Regulatory Commission, (May 21, 1984).

5. M.R. Kuhlman, D.J. Lehmicke and R.O. Meyer, "CORSOR Users' Manual," Battelle Columbus Laboratories, NUREG/CR-4173, BMI-2122, March 1985.

6. R.A. Lorenz, E.C. Beahm and R.P. Wichner, "Review of Tellurium Release Rates from LWR Fuel Elements Under Accident Conditions," Proceedings of the International Meeting on Light Water Reactor Severe Accident Evaluation, August 28-September 1, 1983, pg. 4.4-1, American Nuclear Society Order 700085, ISBN 0-89448-1112-6.

7. L.T. Ritchie, J.D. Johnson and R.M. Blond, Calculations of Reactor Accident Consequences Version 2, CRAC2: Computer Code, User's Guide, prepared by Sandia National Laboratories for the U.S. Nuclear Regulatory Commission, NUREG/CR-2326 (SAND81-1994), February 1983.

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Table 4.1 Comparison of Radioactive Inventories of Equilibrium Core with Spent Fuel Assemblies for Selected Isotopes (Millstone 1)

Isotope	Equilibrium Core	Spent Fuel Pool[a] (time after last discharge) 30-days	90-days	1 year
	(Total Radioactivity, Curies)			
Co 58	8.81E+4	2.29E+4	1.26E+4	8.54E+2
Co 60	1.64E+5	3.72E+5	3.15E+5	2.85E+5
Kr 85	5.35E+5	1.41E+6	1.39E+6	1.33E+6
Rb 86	6.22E+4	1.01E+4	1.05E+3	3.84E-2
Sr 89	4.71E+7	8.39E+6	3.63E+6	8.33E+4
Sr 90	4.25E+6	1.42E+7	1.42E+7	1.39E+7
Y 90	4.37E+6	1.43E+7	1.42E+7	1.39E+7
Y 91	6.06E+7	1.18E+7	5.75E+6	2.21E+5
Zr 95	8.70E+7	1.94E+7	1.00E+7	5.10E+5
Nb 95	8.91E+7	2.54E+7	1.70E+7	1.11E+6
Mo 99	8.78E+7	1.49E+4	3.12E-3	neg.[b]
Tc 99m	7.69E+7	1.43E+4	3.01E-3	neg.[b]
Ru 103	7.23E+7	1.53E+7	5.21E+6	4.07E+4
Ru 106	2.48E+7	1.72E+7	1.53E+7	9.13E+6
Rh 106	2.63E+7	1.72E+7	1.53E+7	9.13E+6
Sb 125	9.07E+5	1.19E+6	1.14E+6	9.48E+5
Sb 127	4.97E+6	8.21E+3	1.39E-1	neg.[b]
Te 125m	1.93E+5	2.84E+5	2.76E+5	2.31E+5
Te 127	4.92E+6	2.21E+5	1.45E+5	2.52E+4
Te 127m	6.61E+5	2.18E+5	1.48E+5	2.57E+4
Te 129	1.49E+7	2.74E+5	7.79E+4	2.68E+2
Te 129m	2.24E+6	4.21E+5	1.20E+5	4.12E+2
Te 132	6.72E+7	3.74E+4	8.64E-2	neg.[b]
I 129	1.75E+0	7.15E+0	7.15E+0	7.15E+0
I 131	4.74E+7	1.22E+6	6.35E+3	neg.[b]
I 132	6.83E+7	3.85E+4	8.90E-2	neg.[b]
Xe 133	9.72E+7	7.29E+5	2.30E+2	neg.[b]
Cs 134	6.10E+6	7.90E+6	7.47E+6	5.80E+6
Cs 136	2.10E+6	2.05E+5	8.13E+3	3.91E-3
Cs 137	5.84E+6	2.02E+7	2.01E+7	1.97E+7
Ba 137m	5.53E+6	1.91E+7	1.90E+7	1.87E+7
Ba 140	8.36E+7	5.19E+6	1.90E+5	6.41E-2
La 140	8.54E+7	5.97E+6	2.19E+5	7.37E-2
Ce 141	7.94E+7	1.32E+7	3.61E+6	1.03E+4
Ce 144	6.05E+7	2.64E+7	2.27E+7	1.16E+7
Pr 143	7.37E+7	5.44E+6	2.41E+5	1.90E-1
Pr 144	6.08E+7	2.64E+7	2.27E+7	1.16E+7
Nd 147	3.16E+7	1.54E+6	3.36E+4	1.10E-3
Sm 151	2.44E+4	8.22E+4	8.21E+4	8.16E+4
Eu 154	4.61E+5	1.34E+6	1.32E+6	1.25E+6
Eu 156	5.61E+6	8.26E+5	5.10E+4	1.80E-1
Np 239	9.98E+8	5.59E+4	2.88E+3	2.88E+3
Pu 238	9.33E+4	4.51E+5	4.53E+5	4.54E+5
Pu 239	2.49E+4	8.89E+4	8.89E+4	8.89E+4
Pu 240	3.14E+4	1.30E+5	1.30E+5	1.30E+5
Pu 241	7.19E+6	2.29E+7	2.27E+7	2.19E+7
Am 241	8.86E+5	2.88E+5	2.94E+5	3.21E+5
Cm 242	2.09E+6	1.45E+6	1.12E+6	3.50E+5
Cm 244	6.72E+4	2.27E+5	2.25E+5	2.19E+5

[a]Spent fuel pool inventory includes discharges from 11 refuelings covering the period from August 1972 through the projected refueling of April 1987.

[b]neg. = less than 10^{-3} Curies.

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Table 4.2 Estimated Radionuclide Release Fraction During a Spent Fuel Pool Accident Resulting in Complete Destruction of Cladding (Cases 1 and 2)

Chemical Family	Element or Isotope	Release Fraction[a] Value Used	Release Fraction[a] Uncertainty Range
Noble gases	Kr, Xe	1.00	0
Halogens	I-129, I-131	1.00	0.5-1.0
Alkali Metals	Cs, (Ba-137m) Rb	1.00	0.1-1.0
Chalcogens	Te, (I-132)	0.02	.002-.02
Alkali Earths	Sr, (Y-90), Ba (in fuel)	2×10^{-3}	10^{-4}-10^{-2}
	Sr, Y-91 (in clad)	1.00	0.5-1.0
Transition Elements	Co-58 (assembly hardware)	0.10	0.1-1.0
	Co-60 (assembly hardware)[b]	0.12	0.1-1.0
	Y-91 (assembly hardware)	0.10	0.1-1.0
	Nb-95, Zr-95 (in fuel)	0.01	10^{-3}-10^{-1}
	Nb-95, Zr-95 (in clad)	1.00	0.5-1.0
Miscellaneous	Mo-99	1×10^{-6}	10^{-8}-10^{-5}
	Ru-106	2×10^{-5}	10^{-6}-10^{-4}
	Sb-125	1.00	0.5-1.0
Lanthanides	La, Ce, Pr, Nd, Sm, Eu	1×10^{-6}	10^{-8}-10^{-5}
Transuranics	Np, Pu, Am, Cm	1×10^{-6}	10^{-8}-10^{-5}

[a]Release fractions of several daughter isotopes are determined by their precursors, e.g., Y-90 by Sr-90, Tc-99m by Mo-99, Rh-106 by Ru-106, I-132 by Te-132, Ba-137m by Cs-137, and La-140 by Ba-140.

[b]Release fraction adjusted to account for a 100% release of the small amount of Co-60 contained in the Zircaloy cladding.

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Table 4.3 Estimated Releases of Radionuclides for Case 1 in Which a Zircaloy Fire Propagates Throughout the Entire Pool Inventory (Worst Case)

Isotope	Time after Last Discharge		
	30-days	90-days	1 year
	(Radioactivity, Curies)		
Co 58	2.74E+3	1.51E+3	1.02E+2
Co 60	4.46E+4	3.78E+4	3.42E+4
Kr 85	1.41E+6	1.39E+6	1.33E+6
Rb 86	1.01E+4	1.05E+5	3.84E-2
Sr 89	1.68E+4	7.26E+3	1.67E+2
Sr 90	2.84E+4	2.84E+4	2.78E+4
Y 90	2.84E+4	2.84E+4	2.78E+4
Y 91	1.18E+6	5.75E+5	2.21E+4
Zr 95	1.63E+6	8.39E+5	4.26E+4
Nb 95	2.13E+6	1.42E+6	9.27E+4
Mo 99	1.49E-2	neg.[a]	neg.[a]
Tc 99m	1.43E-2	neg.[a]	neg.[a]
Ru 103	3.06E+2	1.04E+2	8.14E-1
Ru 106	3.44E+2	3.06E+2	1.83E+2
Rh 106	3.44E+2	3.06E+2	1.83E+2
Sb 125	1.19E+6	1.14E+6	9.48E+5
Sb 127	8.21E+3	1.39E-1	neg.[a]
Te 125m	5.68E+3	5.52E+3	4.62E+3
Te 127	4.42E+3	2.90E+3	5.04E+2
Te 127m	4.36E+3	2.96E+3	5.14E+2
Te 129	5.48E+3	1.56E+3	5.36E+0
Te 129m	8.42E+3	2.40E+3	8.24E+0
Te 132	7.48E+2	1.72E-3	neg.[a]
I 129	7.15E+0	7.15E+0	7.15E+0
I 131	1.22E+6	6.35E+3	neg.[a]
I 132	7.70E+2	1.78E-3	neg.[a]
Xe 133	7.29E+5	2.30E+2	neg.[a]
Cs 134	7.90E+6	7.47E+6	5.80E+6
Cs 136	2.05E+5	8.13E+3	3.91E-3
Cs 137	2.02E+7	2.01E+7	1.97E+7
Ba 137m	1.91E+7	1.90E+7	1.87E+7
Ba 140	1.04E+4	3.80E+2	neg.[a]
La 140	1.19E+4	4.38E+2	neg.[a]
Ce 141	1.32E+1	3.61E+0	1.03E-2
Ce 144	2.64E+1	2.27E+1	1.16E+1
Pr 143	5.44E+0	2.41E-1	neg.[a]
Pr 144	2.64E+1	2.27E+1	1.16E+1
Nd 147	1.54E+0	3.36E-2	neg.[a]
Sm 151	8.22E-2	8.21E-2	8.16E-2
Eu 154	1.34E+0	1.32E+0	1.25E+0
Eu 156	8.26E-1	5.10E-2	neg.[a]
Np 239	5.59E-2	2.88E-3	2.88E-3
Pu 238	4.51E-1	4.53E-1	4.54E-1
Pu 239	8.89E-2	8.89E-2	8.89E-2
Pu 240	1.30E-1	1.30E-1	1.30E-1
Pu 241	2.29E+1	2.27E+1	2.19E+1
Am 241	2.28E-1	2.94E-1	3.21E-1
Cm 242	1.45E+0	1.12E+0	3.50E-1
Cm 244	2.27E-1	2.25E-1	2.19E-1

[a]neg. = less than 10^{-3} Curies.

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Table 4.4 Estimated Releases of Radionuclides for Case 2 In Which Only the Last Discharged Fuel Batch Suffers a Zircaloy Fire

Isotope	Time after Last Discharge		
	30-days	90-days	1 year
	(Radioactivity, Curies)		
Co 58	2.28E+3	1.26E+3	8.49E+1
Co 60	9.17E+3	8.68E+3	8.12E+3
Kr 85	2.39E+5	2.36E+5	2.25E+5
Rb 86	1.01E+4	1.05E+3	3.84E-2
Sr 89	1.79E+4	7.75E+3	1.78E+2
Sr 90	3.84E+3	3.82E+3	3.78E+3
Y 90	3.86E+3	3.84E+3	3.78E+3
Y 91	2.66E+4	1.30E+4	4.99E+2
Zr 95	1.62E+6	8.37E+5	4.25E+4
Nb 95	2.11E+6	1.41E+6	9.24E+4
Mo 99	1.49E-2	neg.[a]	neg.[a]
Tc 99m	1.43E-2	neg.[a]	neg.[a]
Ru 103	3.06E+2	1.04E+2	8.14E-1
Ru 106	2.24E+2	1.99E+2	1.19E+2
Rh 106	2.24E+2	1.99E+2	1.19E+2
Sb 125	4.17E+5	4.00E+5	3.31E+5
Sb 127	8.21E+3	1.39E-1	neg.[a]
Te 125m	1.88E+3	1.88E+3	1.61E+3
Te 127	4.28E+3	2.80E+3	4.86E+2
Te 127m	4.20E+3	2.86E+3	4.96E+2
Te 129	5.48E+3	1.56E+3	5.36E+0
Te 129m	8.42E+3	2.40E+3	8.24E+0
Te 132	7.48E+2	1.73E-3	neg.[a]
I 129	8.84E-1	8.86E-1	8.86E-1
I 131	1.22E+6	6.35E+3	neg.[a]
I 132	7.70E+2	1.78E-3	neg.[a]
Xe 133	7.29E+5	2.30E+2	neg.
Cs 134	3.53E+6	3.34E+6	2.59E+6
Cs 136	2.05E+5	8.13E+3	3.91E-3
Cs 137	2.83E+6	2.81E+6	2.77E+6
Ba 137m	2.67E+6	2.66E+6	2.62E+6
Ba 140	1.04E+5	3.80E+3	1.28E-3
La 140	1.19E+4	4.38E+2	neg.[a]
Ce 141	1.32E+1	3.61E+0	1.03E-2
Ce 144	1.91E+1	1.65E+1	8.43E+0
Pr 143	5.44E+0	2.41E-1	neg.[a]
Pr 144	1.91E+1	1.65E+1	8.43E+0
Nd 147	1.34E+0	3.36E-2	neg.[a]
Sm 151	9.31E-3	9.30E-3	9.25E-3
Eu 154	2.89E-1	2.85E-1	2.69E-1
Eu 156	8.37E-1	5.82E-2	neg.[a]
Np 239	5.36E-2	neg.[a]	neg.[a]
Pu 238	6.75E-2	6.87E-2	7.18E-2
Pu 239	9.28E-3	9.28E-3	9.28E-3
Pu 240	1.35E-2	1.55E-2	1.55E-2
Pu 241	3.73E+0	3.70E+0	3.56E+0
Am 241	6.01E-3	7.00E-3	1.14E-2
Cm 242	1.31E+0	1.01E+0	3.16E-1
Cm 244	5.88E-2	5.84E-2	5.68E-2

[a]neg. = less than 10^{-3} Curies.

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Table 4.5 Estimated Releases of Radionuclides for Cases 3 and 4 in Which Low-Temperature Cladding Failures Occur

Isotope	Case 3[a]	Case 4[b]
	(Radioactivity, Curies)	
Kr 85	6.65E+5	2.39E+5
I 129	3.58E+0	8.84E-1
I 131	neg.[c]	1.22E+6
I 132	neg.[c]	7.70E+2
Xe 133	neg.[c]	7.29E+5

[a]Case 3 assumes:
1. last fuel discharge has decayed for 1 year.
2. no Zircaloy fire occurs.
3. 50% of the fuel rods develop leaks.
4. 100% release of noble gases and halogens from leaking fuel rods.

[b]Case 4 assumes:
1. last fuel batch discharged has decayed for 30 days.
2. no Zircaloy fire occurs.
3. 100% of fuel rods in last discharge develop leaks.
4. 100% release of noble gases and halogens from leaking fuel rods.

[c]neg. = less than 10^{-3} Curies.

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Table 4.6 Comparison of Radioactive Inventories of Equilibrium Core with Spent Fuel Assemblies for Selected Isotopes (Ginna)

Isotope	Equilibrium Core	Spent Fuel Pool[a] (time after last discharge) 30-days	90-days	1 year
	(Total Radioactivity, Curies)			
Co 58	3.57E+5	5.93E+4	3.26E+4	2.21E+3
Co 60	3.20E+5	5.97E+5	5.84E+5	5.29E+5
Kr 85	3.73E+5	9.84E+5	9.74E+5	9.27E+5
Rb 86	6.53E+5	7.22E+3	7.48E+2	2.74E-2
Sr 89	3.55E+7	3.53E+6	1.53E+6	3.50E+4
Sr 90	2.95E+6	1.02E+7	1.01E+7	9.95E+6
Y 90	3.15E+6	1.02E+7	1.01E+7	9.95E+6
Y 91	4.57E+7	5.11E+6	2.48E+6	9.54E+4
Zr 95	6.41E+7	8.64E+6	4.46E+6	2.27E+5
Nb 95	6.34E+7	1.12E+7	7.51E+6	4.93E+5
Mo 99	6.83E+7	7.03E+3	1.48E-3	neg.[b]
Tc 99m	5.89E+7	6.77E+3	1.42E-3	neg.[b]
Ru 103	5.85E+7	7.86E+6	2.88E+6	2.09E+4
Ru 106	1.95E+7	1.09E+7	9.71E+6	5.78E+6
Rh 106	2.15E+7	1.09E+7	9.71E+6	5.78E+6
Sb 125	6.04E+5	7.11E+5	6.82E+5	5.65E+5
Sb 127	4.12E+6	4.33E+3	7.35E-2	neg.[b]
Te 125m	1.27E+5	1.70E+5	1.65E+5	1.37E+5
Te 127	4.05E+6	1.19E+5	7.79E+4	1.36E+4
Te 127m	5.19E+5	1.17E+5	7.95E+4	1.38E+4
Te 129	1.21E+7	1.38E+5	3.93E+4	1.35E+2
Te 129m	1.80E+6	2.12E+5	6.03E+4	2.07E+2
Te 132	5.33E+7	1.83E+4	4.23E-2	neg.[b]
I 129	1.27E+0	5.32E+0	5.32E+0	5.32E+0
I 131	3.76E+7	6.00E+5	3.12E+5	neg.[b]
I 132	5.42E+7	1.89E+4	4.36E-2	neg.[b]
Xe 133	7.64E+7	3.52E+5	1.11E+2	neg.[b]
Cs 134	5.82E+6	6.35E+6	6.00E+6	4.66E+6
Cs 136	1.87E+6	1.26E+5	4.99E+3	2.40E-3
Cs 137	4.21E+6	1.48E+7	1.47E+7	1.44E+7
Ba 137m	4.00E+6	1.40E+7	1.39E+7	1.37E+7
Ba 140	6.55E+7	2.47E+6	9.07E+4	3.05E-2
La 140	6.74E+7	2.85E+6	1.04E+5	3.51E-2
Ce 141	6.28E+7	6.34E+6	1.72E+6	4.91E+3
Ce 144	4.24E+7	1.38E+7	1.19E+7	6.09E+6
Pr 143	5.71E+7	2.54E+6	1.12E+5	8.86E-2
Pr 144	4.27E+7	1.38E+7	1.19E+7	6.09E+6
Nd 147	2.48E+7	7.42E+5	1.62E+4	neg.[b]
Sm 151	1.42E+4	5.14E+4	5.13E+4	5.10E+4
Eu 154	4.09E+5	1.09E+6	1.07E+6	1.01E+6
Eu 156	7.22E+6	7.58E+5	4.68E+4	1.66E-1
Np 239	7.81E+8	3.02E+4	3.26E+3	3.26E+3
Pu 238	1.01E+5	4.46E+5	4.46E+5	4.46E+5
Pu 239	1.35E+4	5.25E+4	5.25E+4	5.25E+4
Pu 240	2.02E+4	8.60E+4	8.60E+4	8.61E+4
Pu 241	4.85E+6	1.52E+7	1.51E+7	1.46E+7
Am 241	4.99E+3	2.10E+5	2.14E+5	2.32E+5
Cm 242	1.91E+6	9.33E+5	7.20E+5	2.25E+5
Cm 244	1.25E+5	3.59E+5	3.56E+5	3.46E+5

[a]Spent fuel pool inventory includes discharges from 15 refuelings covering the period from April 1983 through the projected refueling of April 1987.

[b]neg. = less than 10^{-3} Curies.

Table 4.7 CRAC2 Results for Various Releases Corresponding to Postulated Spent Fuel Pool Accidents with Total Loss of Pool Water

	Case Description	Whole Body Dose (Man-rem)	Interdiction Area (sq. miles)
1A.	Total inventory 30 days after discharge 50 mile radial zone	2.6×10^6	224
1B.	Total inventory 90 days after discharge 50 mile radial zone	2.6×10^6	215
1C.*	Total inventory 30days after discharge 500 mile radial zone	7.1×10^7	224
2A.	Last fuel discharge 90 days after discharge 50 mile radial zone (maximum release fraction)	2.3×10^6	44
2B.	Last fuel discharge 90 days after discharge 50 mile radial zone (mimimum release fraction)	1.1×10^6	4
3.	50% of all fuel rods leak 1 year after discharge 50 mile radial zone	4.0	0.0

*Note that the consequence calculations in NUREG-1150 are based on a 50 mile radial zone. Case 1C is given as a sensitivity result.

5. RISK PROFILE

The likelihood and consequences of various spent fuel pool accidents has been estimated in the previous sections. The risk is summarized in Table 5.1. As previously mentioned, the exposure results are tied to the health physics assumptions regarding decontamination and maximum allowable exposure. Thus the land interdiction area is included in Table 5.1 as a more meaningful representation of severity. The uncertainty in each of these risk indices is quite large and is due principally to the uncertainty in the fragility of the pools and the uncertainty in the seismic hazard, but there is also a significant uncertainty in the human error contribution to the cask drop frequency and the fission product release estimates.

Note that the risk results are calculated for two older plants with full fuel pools which were identified as being potentially vulnerable to spent fuel pool accidents. The present risk estimates are not expected to be applicable to more recent plants for which the fuel pools have been designed to more stringent seismic criteria (and can therefore be expected to be less susceptible to seismic failure). There is a potential for larger fission product inventories when some of the newer dual unit plants begin to fill up the pools. However, the seismic margin in the newer designs appears to contribute to an overall risk reduction.

5.1 Failure Frequency Estimates

5.1.1 Spent Fuel Pool Failure Probability

The likelihood of the various postulated spent fuel pool accidents was developed in Section 2 and summarized in Table 2.9. The major contributors to the estimated accident frequency are:

1. Cask drop accidents,
2. Seismic induced pool failure,
3. Loss of pool cooling, and
4. Pneumatic seal failure.

Note that all of these potential accidents are plant specific and their frequency will vary widely from plant to plant. In particular, most BWR's do not have pneumatic seals so their failure frequency is zero. At least one dual unit BWR (Hatch) has its fuel pool outside the reactor building and uses pneumatic seals during refueling operations.

5.1.2 Spent Fuel Failure Likelihood

Previous investigations[1,2] of spent fuel behavior after a loss of pool integrity accident focused on the conditions necessary to initiate cladding "fires" after a spent fuel pool has drained. The present project has reevaluated these conditions using the SFUEL code[2] developed by SNL. The likelihood of such cladding fires has been assessed in Section 3. For a PWR with high density storage racks, the conditional probability of a clad fire given a pool drainage event, was found to be about 1.0 while for a BWR with low density storage racks the probability of a clad fire was found to be about 0.25. The improved air circulation and the lower power density contribute to the improved cooling (and reduced likelihood of fire) for the BWR storage racks.

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5.2 Conclusions Regarding Risk

The overall risk due to beyond design basis accidents in spent fuel pools for the PWR and BWR plants is shown in Table 5.1. The unique character of such an accident (substantial releases of long lived isotopes) makes it difficult to compare to reactor core melt accidents. The exposure calculations are driven by assumptions in the CRAC modeling and the results are not sensitive to the severity of the accident.

The uncertainty in risk in terms of person-rem/Ry is driven principally by the uncertainty in the calculation of initiating event frequency for events which completely drain the spent fuel pool. The uncertainty in event frequency is, in turn, driven by the uncertainty in the seismic hazard and fragility along with the uncertainty in human error for cask drop accidents.

5.3 References for Section 5

1. A.S. Benjamin, D.J. McCloskey, D.A. Powers, S.A. Dupree, "Spent Fuel Heat-up Following Loss of Water During Storage," NUREG/CR-0649, March 1979.

2. N.A. Pisano, F. Best, A.S. Benjamin, K.T. Stalker, "The Potential for Propagation of a Self-Sustaining Zirconium Oxidation Following Loss of Water in a Spent Fuel Storage Pool," Draft Report, January 1984.

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Table 5.1 Estimated Risk for the Two Spent Fuel Pools from the Two Dominant Contributors

Accident Initiator	Spent Fuel Pool Fire Probability/Ry	Health Risk[1] (Man-rem/Ry)	Interdiction[1] Risk (Sq. Mi./Ry)
Seismic induced PWR pool failure	2.6×10^{-4}-1.6×10^{-10}	600-Neg.*	.011-Neg.
Seismic induced BWR pool failure	6.5×10^{-5}-4×10^{-11}	156-Neg.	.003-Neg.
Cask drop[2] induced PWR pool failure	3×10^{-5}-3×10^{-12}	70-Neg.	.001-Neg.
Cask drop[2] induced BWR pool failure	8×10^{-6}-8×10^{-13}	20-Neg.	4×10^{-4}-Neg.

*Neg. - Negligible.

[1]The upper end of the risk ranges assumes no fire propagation from the last fuel discharge to older fuel. However, the fission products in the last fuel discharge were assumed to be released during the fire with no fission product decontamination on structures (Case 2A).

[2]After removal of accumulated inventory resumes. Presently, most plants are accumulating spent fuel in the pool without shipping to permanent storage. (Note that many new plants have pool configurations and administrative procedures which would preclude this failure mode.)

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6. CONSIDERATION OF RISK REDUCTION MEASURES

Due to diversity in the nature of initiating events for beyond design basis accidents in spent fuel pools, there appear to be several possible ways to reduce the risks. It must be emphasized that each of the contributors to risk are plant specific and one or more of the risk significant sequences identified in Section 5 may not be important at other plant sites. The following sections discuss the advantage and disadvantages of a number of proposed risk reduction strategies. A cost benefit analysis has not been performed. Rather, the phenomenological insights, developed during the investigation, have been used to generate the lists of possible risk reduction measures provided below.

6.1 Risk Prevention

1. Reduction of Stored Radioactive Inventory - Most of the consequences of a release of radioactivity from a pool accident is associated with the large inventory of isotopes of intermediate halflives, e.g., Cs-137, Sr-90. The potential release increases approximately in proportion to the number of fuel assemblies in the storage inventory. One obvious measure for risk reduction is to transfer part of the inventory to alternative storage locations (e.g., see Ref. 1).

2. Air Circulation - The one universal prevention measure is to promote air cooling in the event of loss of water cooling of the spent fuel. The new high density fuel storage racks restrict air flow and make even old spent fuel (one to two years) susceptible to heat-up and self-sustaining oxidation. The older style fuel baskets with large inlet holes (3 inch diameter or more per assembly) allow much freer air circulation. If all recently discharged fuel (less than two years) is kept in low density fuel baskets and they are separated from the wall and the older fuel by a one foot gap then the likelihood of self-sustaining oxidation would be reduced by a factor of 5 or more compared to the high density storage configuration.

3. Additional Cooling Systems - Although loss-of-pool cooling appears to be risk significant, an additional cooling system is unlikely to be cost beneficial (unless the cooling system for a specific plant was substantially more unreliable than the two current systems). An additional cooling system would not affect the risk from pool failure events (seismic or cask drop accidents). Thus the net risk reduction would be minimal unless loss-of-cooling were the dominant event.

4. Improved Procedures and Equipment - The likelihood of cask drop accidents can be reduced by improving procedures, administrative controls and/or installing more reliable equipment as suggested in the heavy load drop investigation.[3] Improvements in operator training, heavy load procedures and equipment, which is recommended as part of the resolution for Generic Issue A-36, is estimated to reduce this failure mode by at least a factor of 1000. However, none of these improvements would reduce the risk from the other dominant sequences. Thus the net risk reduction would have to be quantified on a plant specific basis.

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5. Plant-Specific Risk Evaluation - This option provides for a limited risk evaluation before spent fuel shipment is begun at each site. A key piece of such an evaluation would be a structural analysis of the pool response to the loading from a dropped cask or a seismic event.

6.2 Accident Mitigation

1. Post-Accident Spray - Water spray has the potential to terminate the progression of a spent fuel pool accident whether or not the pool is intact. However, large quantities of water must be available (it would be necessary to continue spraying until the pool could be repaired and reflooded) and the equipment would have to be seismically qualified to a higher ground motion (g) level than the pool structure (in order for the sprays to have a high likelihood of surviving). Some pools may have fire sprays available in the spent fuel pool building. For those plants without sprays available, it seems unlikely that the expense of a new safety grade spray system could be justified considering the large uncertainty in the risk. Temporary fire hoses were suggested by Benjamin et al.,[2] but the radiation levels would make such ad hoc measures extremely difficult. Furthermore, if the spray is not initiated before the rods reach 900 C or there is insufficient flow, the water may aggravate the reaction by providing additional oxidation potential. (The steam/Zircaloy reaction is also highly exothermic.)

2. Filtering - For those plants with a standby gas treatment system available, operation of the system has the potential to substantially reduce the fission product release from the building. However, the high temperatures and large aerosol production rate would tend to rapidly degrade the effectiveness of the system. The performance of such a filtering system would be difficult to characterize under fuel pool accident conditions. It is unlikely to be cost effective to install a new system large enough to handle the worst case spent fuel pool accident scenarios.

3. Fuel Fire Retardation Modules - Certain materials are known to retard fuel fires, as was demonstrated during the 1986 Chernobyl accident (sand, boron, etc.). Such materials stored in large quantities on site, with a special emergency transport system (helicopter, robots, etc.) could allow for a viable spent fuel pool fire mitigation system.

6.3 Conclusions Regarding Preventive and Mitigative Measures

For those plants which have a significant spent fuel pool risk, the one preventive measure which appears to have a substantial effect on risk (a risk reduction of 5 or more) is to maintain recently discharged fuel in low density storage racks that are isolated from the rest of the fuel racks by a foot or more of space (to provide free air circulation). However, there may be plant specific features which make a substantial difference in the order of the dominant contributors to risk. Therefore plant specific risk evaluations should be performed before any changes are implemented at a given plant.

6.4 References for Section 6

1. D.D. Orvis, C. Johnson, and R. Jones, "Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment," prepared for the Electric Power Research Institute by the NUS Corporation, EPRI NP-3365, February 1984.

2. A.S. Benjamin, D.J. McCloskey, D.A. Powers, S.A. Dupree, "Spent Fuel Heat-up Following Loss of Water During Storage," NUREG/CR-0649, March 1979.

3. H. George, "Control of Heavy Loads at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, NUREG-0612, January 1980.

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APPENDIX A

RADIOACTIVE INVENTORIES

A.1 INTRODUCTION

The radionuclide inventories contained in various spent fuel batches has been calculated for two older plants, Millstone-1 (BWR) and Ginna (PWR). The purpose of this appendix is to describe the methods used to simulate the operating history of the two plants and to summarize the calculated radioactive inventories contained in the fuel assemblies stored in the spent fuel basins.

A.2 SIMULATION OF OPERATING HISTORIES

A.2.1 Thermal Energy Production vs Time

The operating history of each plant was reconstructed from several sources. The early history, prior to December 1, 1975 was reconstructed from monthly summaries contained in Refs. 1-3. Data for the period December 1, 1975 through April 30, 1986 were taken from Ref. 4. Data from May 1, 1986 to April 1, 1987 were extrapolated, based on recent average capacity factors and scheduled shutdowns.

During each operating cycle (the period between successive refuelings), the average thermal power was calculated from the total thermal energy produced during the cycle. No attempt was made to model variations in power levels during an operating period. (Fluctuations in the monthly energy production are illustrated in Fig. A.1.)

A.2.2 Fuel Burnup Calculations

The number of fuel assemblies discharged at each refueling and their specific burnup was obtained from a data base maintained by R.A. Libby of Pacific Northwest Laboratories (PNL) for the U.S. Department of Energy.[5] It should be noted that the inventory of spent fuel assemblies stored in the spent fuel basins at various points in time listed in the Libby data base differ from the data listed in Ref. 4. It is apparent from the operating histories that the data in the earlier volumes of Ref. 4 are less accurate.

In general, the burnups listed in the Libby data base differ by a few percent from the burnups calculated by the methods described in the following paragraphs. These discrepancies do not have significant effects on the overall inventories of radionuclides, but only on the distribution of the inventories among the older fuel batches.

In order to model the burnup of the various discharged batches of spent fuel, the following method was used. It was assumed that all fuel assemblies in the core during a given operating cycle provided the average specific power, i.e.,

$$(MW_{th}/MT)_i = (MW_{th}D)_i/D_i(MT)_{core} ,$$

where for operating cycle, i, MW_{th}/MT is the average specific power per metric tonne of initial heavy metal, $(MW_{th}D)_i$ is the total thermal energy produced in D_i days of the cycle, i, and MT_{core} is the metric tonnes of initial heavy metal in the core.

The average specific burnup for each fuel batch, j, at discharge was calculated from the formula,

$$(MW_{th}D/MT)_j = \sum_i (MW_{th}/MT)_i D_i \ ,$$

where $\sum_i$ is the summation over the several operating cycles, i, that the fuel was in the reactor. (As noted below, ORIGEN2 also calculates the specific burnup which provides a check on internal consistency of the data).

The total burnup in the discharged fuel plus the burnup of assemblies remaining in the core at the time of the April 1, 1987 refueling equaled the total thermal energy production over the preceding history of the plant (e.g., see Table A.4).

A.2.3 Calculation of Radioactive Inventories

The average radionuclide content in each metric tonne of discharged fuel was calculated using the ORIGEN2 Computer Code.[6] The calculations treated the reactor core as a homogeneous body operating at an average specific power. Account is taken of radionuclide decay during and following irradiation, decay chains, and successive neutron captures.

The BNL version of ORIGEN2 was benchmarked against the version in use at Oak Ridge National Laboratory by calculating an identical case, which yielded identical results.[7]

The results obtained from an ORIGEN2 calculation are slightly sensitive to the size of the time steps used in the irradiation calculation. Several preliminary calculations were made to select an appropriate set of time steps for which the sensitivity was negligible. (Shorter time steps give higher precision results, but at the expense of increased computer time. The criterion adopted was that the time-step sensitivity be less than 0.1% in the calculated concentration of several key nuclides.)

In a mature operating nuclear power plant fuel management strategies are complicated (e.g., see Ref. 8). Most fuel assemblies remain in the core for several operating cycles and are often shifted in location during refueling so as to optimize burnup. Also, U-235 enrichment is varied. ORIGEN2 as used at BNL did not take account of such detail, nor of the axial and radial distribution of the power density. Thus, the radioactivity calculated for a particular assembly would not correspond exactly to an actual assembly. Nevertheless, the total calculated radioactivity in a discharged batch closely approximates the total in a real batch (in so far as the precision of ORIGEN2 allows).

The calculations do take account of the irradiation times in each operating cycle and the decay that occurs during shutdowns for refueling or prolonged shutdowns for maintenance and repair.

As used at BNL, the input for each irradiation cycle is the average specific power and the length of the cycle. ORIGEN2 calculates the total average burnup of each fuel batch over the irradiation cycles during which it was in the core. This calculated burnup was cross checked against "hand" calculations for each batch, the "hand" calculations being based on the operating history (see Section A.2.2).

The input for ORIGEN2 requires the specification of the elements contained in the fuel including trace impurities, the U-235 enrichment and the composition and amount of alloys used in the fuel cladding and assembly hardware. For each plant, BWR and PWR, only a single fuel and assembly composition was modeled which is typical of fuel of recent vintage for the respective reactors. Data for the fuel models were taken from Reference 9.

The output of ORIGEN2 includes isotopic concentrations (of stable as well as radioactive isotopes), activity of radionuclides, and thermal power production of each radionuclide. These are given at specified decay times for activation products (in cladding, hardware and trace elements in the fuel pellets), fission products and actinides.

The BNL calculations were made for each fuel batch from the date of the end of irradiation to the projected dates of May 1, 1987, July 1, 1987, October 1, 1987 and April 1, 1988.

A.3 DATA FOR MILLSTONE 1

A.3.1 Reactor and Fuel Cycle Parameters

Table A.1 summarizes several of the major reactor characteristics and fuel cycle parameters for Millstone 1.

A.3.2 History of Operations

Several milestones in the operation of Millstone-1 are summarized in Table A.2. Monthly gross thermal energy production from 1976 through 1984 is plotted in Fig. A.1. During the first 10 years of operation the plant experienced two prolonged outages, i.e. Sept. 1972 to March 1973 (198 days) and October 1980 to June 1981 (254 days). Otherwise the refueling/maintenance outages have ranged from 35 to 76 days in duration averaging about 57 days.

A more detailed narrative of the plant operating history from 1970 through 1981 appears in Ref. 10, Appendix F, pp. F-31 through F-70. The only unusual experience with fuel cladding failures that has been noted occurred in 1974 when some 25 assemblies were found to have leaking fuel elements which forced a temporary power derating to stay within off-gas release limits.[11] Since mid-1981, the plant has operated with nearly 100% unit service factor except for scheduled refueling outages.[4]

There have been 10 refueling campaigns since beginning of commercial operations on March 1, 1971 (see Table A.3). The next scheduled refueling will

be about April 1987. During the first 10 years, refueling occurred at somewhat irregular intervals, being dictated by unscheduled forced outages. Since 1981, refueling has been scheduled for approximately 18 month intervals, occurring in April or September. During the lifetime of the plant the average fuel burnup has generally increased from about 20,000 MWD/MT in 1972 to about 28,000 MWD/MT at present.

A.3.3 BWR Fuel Assembly Model Used in ORIGEN2 Calculations

A nominal BWR fuel element has been modeled, based on data presented in Ref. 9. This is an 8x8 element assembly of 2.75% U-235 enrichment, containing 1.5873 kg of gadolinium burnable poison per metric tonne of uranium. The fuel cladding is Zircaloy-2. Other alloys present in the fuel assembly hardware include Zircaloy-4, Inconel X-750, SS302 and SS304. The alloy contents of the assembly hardware are included with weighting factors to take account of the axial variation of neutron flux which results in lower neutron activation at the ends of the assemblies. In addition to the fuel, the cladding and the assembly hardware, an allowance was made for the presence of "crud" composed of Fe, Co, and Ni on the outer surfaces of the cladding and assembly hardware.

A.3.4 Calculated Radioactive Inventories

The calculated inventories of selected radionuclides* are listed in Table A.5 for the reactor core at the end of operating cycle number 11 projected to be on April 1, 1987. Also listed are the inventories in the spent fuel basin on May 1 and July 1, 1987 and April 1, 1988 assuming that 167 assemblies will be discharged in the April 1987 refueling.

It should be noted that many of the isotopes that are of considerable importance in a core melt accident are those of short half-lives which are no longer present in the spent fuel after a few days of decay, e.g., Rb-91, Rb-93, Sr-93, Sr-95, Y-94, Y-95, Tc-104, I-134, I-135, I-136, Cs-138, Cs-140. On the other hand, the spent fuel inventory contains much larger quantities of several long-lived isotopes than does the equilibrium core. Noteworthy among these are H-3, C-14, Sr-90 (Y-90), I-129, Cs-137, Ba-137m, Eu-154, Pu-239, Pu-240, Pu-241, Am-241, and Cm-244.

Table A.6 gives a comparison of the radionuclide inventories in the last fuel batch to be discharged with the summation of the inventories contained in the ten batches discharged in the period from 1972 through 1985.

A.3.5 Decay Heat

Table A.7 summarizes the decay thermal production in the various discharged batches. The data shown is for the whole batch, i.e., the specific thermal power (kilowatts per metric tonne) multiplied by the metric tonnes in the batch.

Table A.8 summarizes the fraction of the decay heat contributed by various isotopes. The main contributors change with decay time, e.g., in the

*The selection of radionuclides was based on several criteria including potential for biological concern, thermal power, and total curies of activity.

oldest fuel (batches 1, 2, etc.) the largest contributors are Y-90 and Ba-137m, whereas the last discharged batch 11 is dominated by Cs-134, Rh-106, and Pr-144. The actinides are relatively small contributors.

A.4 DATA FOR GINNA

A.4.1 Reactor and Fuel Cycle Parameters

Table A.9 summarizes several of the major reactor characteristics and fuel cycle parameters for Ginna.

A.4.2 History of Operations

Several milestones in the operation of Ginna are summarized in Table A.10. A narrative of the operating history from 1969 through 1979 can be found in Ref. 12, Appendix F.

Reconstruction of the refueling history during the early years of operation has been difficult using data readily accessible to BNL Staff (direct access to the Licensee for information was precluded). Table A.11 lists the refueling data used by BNL for the ORIGEN2 calculations, which were carried out in 1985.

Subsequently, additional information has been located that would permit a revision of the data in Table A.11, but repeating the ORIGEN2 calculations did not seem worthwhile since only minor changes in the spent fuel radioactive inventories would have resulted. At the time Table A.11 was constructed, no data on the first refueling in February, 1971 was available. Also, some 84 fuel assemblies from early refuelings could not be accounted for. Later, it was learned that 81 assemblies had been shipped for reprocessing at the West Valley facility. These apparently were returned in 1985 to Ginna for storage in the spent fuel pool.[13]

At the time of the April 1972 refueling, cladding distortions due to fuel densification was discovered and 61 assemblies were replaced (Ref. 12, pg. F-56). Thus, the entry in Table A.11 for the second discharge is incorrect.

The total burnup not accounted for in the ORIGEN2 calculations amounts to 4.2% of the total thermal energy production from 1969 through April 1, 1987. The missing 4.2% burnup is for fuel discharged on or before April 1972.

A.4.3 PWR Fuel Assembly Model Used in ORIGEN2 Calculations

A nominal PWR fuel element has been modeled based on data presented in Ref. 9. This is a 17x17 element assembly (264 fuel elements per assembly) of 3.2% U-235 enrichment containing 461.4 kg of uranium. The cladding is Zircaloy-4. Other alloys present in the fuel assembly hardware include Inconel-718, Nicrobraze 50, SS-302 and SS-304. The alloy contents of the assembly hardware are included with weighting factors to take account of the axial variation of the neutron flux which results in lower neutron flux which results in lower neutron activation at the ends of the assemblies. In addition to the fuel, the cladding and the assembly hardware, an allowance was made for the presence of "crud," composed of Cr, Fe, Co and Ni, on the outer surfaces of the cladding and hardware.

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No corrections were made in the ORIGEN2 calculations to account for stainless steel clad fuel that was used in the early history of the plant.

A.4.4 Calculated Radioactive Inventories

The calculated inventories of selected radionuclides* are listed in Table A.12 for the end of operating cycle number 16 projected to be on April 1, 1987. Also listed are the inventories in the spent fuel basin on May 1 and July 1, 1987 and April 1, 1988, assuming that 24 assemblies will be discharged in the April 1987 refueling.

It should be noted that many of the isotopes that are of considerable importance in a core melt accident are those of short half-lives which are no longer present in the spent fuel after a few days of decay, e.g., Rb-91, Rb-93, Sr-93, Sr-95, Y-94, Y-95, Tc-104, I-134, I-135, I-136, Cs-138, Cs-140. On the other hand, the spent fuel inventory contains much larger quantities of several long-lived isotopes than does the equilibrium core. Noteworthy among these are H-3, C-14, Sr-90 (Y-90), I-129, Cs-137, Ba-137m, Eu-154, Pu-239, Pu-240, Pu-241, Am-241, and Cm-244.

Table A.13 gives a comparison of the radionuclide inventories in the last fuel batch to be discharged with the summation of the inventories contained in batches 2-15 discharged between 1976 and 1986.

A.4.5 Decay Heat

Table A.14 summarizes the decay heat production in the various discharged batches. The data shown is for the whole batch, i.e., the specific thermal power (kilowatts per metric tonne) multiplied by the metric tonnes in the batch.

Table A.15 summarizes the fraction of the decay heat contributed by various isotopes. The main contributors change with decay time, e.g., in the oldest fuel (batches 2, 3, etc.) the largest contributors are Y-90 and Ba-137m, whereas the last discharged batch 16 is dominated by Cs-134, Rh-106 and Pr-144. The actinides are relatively small contributors.

*The selection of radionuclides was based on several criteria including potential for biological concern, thermal power and total curies of activity.

A.5 REFERENCES FOR APPENDIX A

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9. A.G. Croff, M.A. Bjerke, G.W. Morrison, and L.M. Petrie, Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code, Oak Ridge National Laboratory, ORNL/TM-6051, September 1978.

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11. U.S. Nuclear Regulatory Commission, Nuclear Power Plant Operating Experience, 1974-1975, NUREG-0227, April 1977, pg. 65.

12. U.S. Nuclear Regulatory Commission, Integrated Plant Safety Assessment Systematic Evaluation Program, R.E. Ginna Nuclear Power Plant, NUREG-0821, December 1982.

13. Rochester Gas and Electric Corporation, "Application for Amendment to Operating License to Amend Appendix A to Increase Spent Fuel Pool Storage Capacity," submitted to NRC April 2, 1984, Docket No.50-244.

Table A.1 Reactor and Fuel Cycle Parameters for Millstone 1 (Sources: Refs. 1-4)

Assemblies in core: 580

Licensed thermal power: 2011 MW_{th} (gross)

Thermal power corresponding to maximum dependable capacity: 2006.5 MW_{th} (gross)

Nominal initial metric tonnes of heavy metal (IMTHM) per assembly: 0.1833 MT

Average refueling cycle interval (since initial commercial operation): 21 to 22 months

Recent refueling cycle interval (since April, 1979): about 18 months

Average number of assemblies per discharge: about 173

Average IMTHM per discharge: about 31.7 MT

Average number of fuel cycles per assembly: about 3.35

Average period of irradiation (including downtime): about 72 months

Authorized Storage Pool Capacity (as of 1985): 2184 assemblies

000311

Table A.2 Summary of Operational Milestones for Millstone 1 (Source: Ref. 4)

Date of Initial Criticality: October 26, 1970

Date of First Electricity Generation: November 29, 1970

Date of Commercial Operation: March 1, 1971

Lifetime Cumulative Data: (January 1, 1971 - March 31, 1986)

Hours, Generator on Line: 100,307.9 hours

Gross Thermal Energy: 184.83×10^6 MWh

Capacity Factor (MDC net): 67.4%

000312

Table A.3 Summary of Spent Fuel Batches in Millstone 1 Storage Basin (With Projections to 1987)

Spent Fuel Batch No.	Date of End of Irradiation	Number of Assemblies	Weight H.M. (MT)	Avg. Burnup (MWD/MT)	Decay[a] Days to 5/1/87 (days)	Cumulative Assemblies in Pool	Cumulative[b] Gross Weight of Spent Fuel in Pool (MT)
1	08/31/72	28	5.132	12686	5356	28	8.95
2	08/31/74	204	38.126	19695	4626	236	75.47
3	09/11/75	144	26.395	26581	4250	380	121.52
4	09/30/76	124	22.729	21290	3865	504	161.18
5	03/10/78	124	22.729	24090	3339	628	200.83
6	04/27/79	148	27.128	24354	2926	776	248.16
7	10/03/80	168	30.794	24998	2394	944	301.89
8	09/11/82	192	35.194	23670	1693	1136	363.29
9	04/12/84	172	31.528	26763	1114	1308	418.30
10[c]	10/01/85	178	32.627	28052	577	1486	475.22
11[c]	04/01/87	167	30.611	29963	30	1653[d]	528.63

[a]Decay days from end of irradiation to 5/1/87.
[b]Gross fuel tonnage in pool includes heavy metal plus cladding and hardware but not including fuel racks. Each assembly contains approximately 0.1833 metric tonnes of heavy metal, 0.0246 tonnes of oxygen (in UO_2) and 0.1119 tonnes of hardware, totaling 0.3198 metric tonnes gross.
[c]Projected data.
[d]The present authorized storage capacity is 2184 assemblies. After the 04/01/87 refueling, the accumulated assemblies plus the 580 assemblies in the core would exceed the authorized storage capacity should a full core discharge be required.

000313

Table A.4 Comparison of Cumulative Gross Thermal Energy Production with Calculated Fuel Burnup from Start of Operations in 1970 to April 1, 1987 (Millstone 1)

Total Cumulative Gross Thermal Energy (MWD x 10^{-3})	Spent Fuel Batch No.	Total Burnup in Batch (MWD x 10^{-3})
	1	65.10
	2	750.88
	3	701.61
	4	483.91
	5	547.54
	6	660.68
	7	769.78
	8	833.05
	9	843.78
	10	915.25
	11	917.21
	12*	612.74
	13*	329.55
Total 8440.25		8440.01

*Burnup in fuel remaining in the core.

Table A.5 Comparison of Radioactive Inventories in Reactor Core and Spent Fuel Basin (Millstone 1). The Assumed Refueling Scenario is Described in Section A.3.4

Isotope	Reactor Core	Spent Fuel Storage Basin[a] 5/1/87	7/1/87	10/1/87	4/1/88
	(Radioactivity, Curies)				
H 3	4.95E+4	1.38E+5	1.37E+5	1.35E+5	1.31E+5
C 14	1.02E+2	4.12E+2	4.12E+2	4.12E+2	4.12E+2
Co 58	8.81E+4	2.29E+4	1.26E+4	5.12E+3	8.54E+2
Co 60	1.64E+5	3.72E+5	3.15E+5	3.04E+5	2.85E+5
Kr 85	5.35E+5	1.41E+6	1.39E+6	1.37E+6	1.33E+6
Rb 86	6.22E+4	1.01E+4	1.05E+3	3.44E+1	3.84E-2
Sr 89	4.71E+7	8.39E+6	3.63E+6	1.03E+6	8.33E+4
Sr 90	4.25E+6	1.42E+7	1.42E+7	1.41E+7	1.39E+7
Y 90	4.37E+6	1.43E+7	1.42E+7	1.41E+7	1.39E+7
Y 91	6.06E+7	1.18E+7	5.75E+6	1.98E+6	2.21E+5
Zr 95	8.70E+7	1.94E+7	1.00E+7	3.70E+6	5.10E+5
Nb 95	8.91E+7	2.54E+7	1.70E+7	7.35E+6	1.11E+6
Mo 99	8.78E+7	1.49E+4	3.12E-3	neg.[b]	neg.[b]
Tc 99m	7.69E+7	1.43E+4	3.01E-3	neg.[b]	neg.[b]
Ru 103	7.23E+7	1.53E+7	5.21E+6	1.03E+6	4.07E+4
Ru 106	2.48E+7	1.72E+7	1.53E+7	1.29E+7	9.13E+6
Rh 106	2.63E+7	1.72E+7	1.53E+7	1.29E+7	9.13E+6
Sb 125	9.07E+5	1.19E+6	1.14E+6	1.07E+6	9.48E+5
Sb 127	4.97E+6	8.21E+3	1.39E-1	neg.[b]	neg.[b]
Te 125m	1.93E+5	2.84E+5	2.76E+5	2.61E+5	2.31E+5
Te 127	4.92E+6	2.21E+5	1.45E+5	8.06E+4	2.52E+4
Te 127m	6.61E+5	2.18E+5	1.48E+5	8.23E+4	2.57E+4
Te 129	1.49E+7	2.74E+5	7.79E+4	1.17E+4	2.68E+2
Te 129m	2.24E+6	4.21E+5	1.20E+5	1.79E+4	4.12E+2
Te 132	6.72E+7	3.74E+4	8.64E-2	neg.[b]	neg.[b]
I 129	1.75E+0	7.15E+0	7.15E+0	7.15E+0	7.15E+0
I 131	4.74E+7	1.22E+6	6.35E+3	2.28E+0	neg.[b]
I 132	6.83E+7	3.85E+4	8.90E-2	neg.[b]	neg.[b]
Xe 133	9.72E+7	7.29E+5	2.30E+2	1.21E-3	neg.[b]
Cs 134	6.10E+6	7.90E+6	7.47E+6	6.86E+6	5.80E+6
Cs 136	2.10E+6	2.05E+5	8.13E+3	6.26E+1	3.91E-3
Cs 137	5.84E+6	2.02E+7	2.01E+7	2.00E+7	1.97E+7
Ba 137m	5.53E+6	1.91E+7	1.90E+7	1.89E+7	1.87E+7
Ba 140	8.36E+7	5.19E+6	1.90E+5	1.30E+3	6.41E-2
La 140	8.54E+7	5.97E+6	2.19E+5	1.50E+3	7.37E-2
Ce 141	7.94E+7	1.32E+7	3.61E+6	5.07E+5	1.03E+4
Ce 144	6.05E+7	2.64E+7	2.27E+7	1.81E+7	1.16E+7
Pr 143	7.37E+7	5.44E+6	2.41E+5	2.19E+3	1.90E-1
Pr 144	6.08E+7	2.64E+7	2.27E+7	1.81E+7	1.16E+7
Nd 147	3.16E+7	1.54E+6	3.36E+4	1.05E+2	1.10E-3
Sm 151	2.44E+4	8.22E+4	8.21E+4	8.19E+4	8.16E+4
Eu 154	4.61E+5	1.34E+6	1.32E+6	1.29E+6	1.25E+6
Eu 156	5.61E+6	8.26E+5	5.10E+4	7.76E+2	1.80E-1
Np 239	9.98E+8	5.59E+4	2.88E+3	2.88E+3	2.88E+3
Pu 238	9.33E+4	4.51E+5	4.53E+5	4.54E+5	4.54E+5
Pu 239	2.49E+4	8.89E+4	8.89E+4	8.89E+4	8.89E+4
Pu 240	3.14E+4	1.30E+5	1.30E+5	1.30E+5	1.30E+5
Pu 241	7.19E+6	2.29E+7	2.27E+7	2.25E+7	2.19E+7
Am 241	8.86E+3	2.88E+5	2.94E+5	3.03E+5	3.21E+5
Cm 242	2.09E+6	1.45E+6	1.12E+6	7.60E+5	3.50E+5
Cm 244	6.72E+4	2.27E+5	2.25E+5	2.23E+5	2.19E+5

[a]Spent fuel pool inventory includes discharges from 11 refuelings covering the period from August 1972 through the projected refueling of April 1987.

[b]neg. = less than 10^{-3} Curies.

Table A.6 Comparison of Radioactive Inventories of Most Recently Discharged Fuel Batch (Batch 11) with Longer Aged Discharged Batches (Batches 1-10) (Millstone 1)

Isotope	Spent Fuel Batch 11[a]				Spent Fuel Batch 1-10[b]			
	5/1/87	7/1/87	10/1/87	4/1/88	5/1/87	7/1/87	10/1/87	4/1/88
	(Radioactivity, Curies)							
H 3	2.27E+4	2.24E+4	2.21E+4	2.15E+4	1.16E+5	1.15E+5	1.13E+5	1.10E+5
C 14	5.18E+1	5.18E+1	5.18E+1	5.18E+1	3.61E+2	3.61E+2	3.61E+2	3.61E+2
Co 58	2.28E+4	1.26E+4	5.10E+3	8.49E+2	1.18E+2	6.48E+1	2.63E+1	4.39E+0
Co 60	7.64E+4	7.48E+4	7.24E+4	6.77E+4	2.45E+5	2.40E+5	2.32E+5	2.17E+5
Kr 85	2.39E+5	2.36E+5	2.32E+5	2.25E+5	1.17E+6	1.16E+6	1.14E+6	1.10E+6
Rb 86	1.01E+4	1.05E+3	3.44E+1	3.84E-2	neg.[c]	neg.	neg.	neg.
Sr 89	8.39E+6	3.63E+6	1.03E+6	8.33E+4	5.39E+3	2.33E+3	6.60E+2	5.35E+1
Sr 90	1.93E+6	1.92E+6	1.91E+6	1.89E+6	1.23E+7	1.23E+7	1.22E+7	1.21E+7
Y 90	1.93E+6	1.92E+6	1.91E+6	1.89E+6	1.23E+7	1.23E+7	1.22E+7	1.21E+7
Y 91	1.18E+7	5.74E+6	1.93E+6	2.21E+5	2.11E+4	1.02E+4	3.44E+3	3.94E+2
Zr 95	1.94E+7	1.00E+7	3.69E+6	5.09E+5	5.90E+4	3.05E+4	1.12E+4	1.55E+3
Nb 95	2.53E+7	1.69E+7	7.33E+6	1.11E+6	1.31E+5	6.76E+4	2.49E+4	3.44E+3
Mo 99	1.49E+4	3.12E-3	neg.	neg.	neg.	neg.	neg.	neg.
Tc 99m	1.43E+4	3.01E-3	neg.	neg.	neg.	neg.	neg.	neg.
Ru 103	1.53E+7	5.21E+6	1.03E+6	4.07E+4	1.09E+3	3.73E+2	7.35E+1	2.91E+0
Ru 106	1.12E+7	9.98E+6	8.40E+6	5.95E+6	5.98E+6	5.30E+6	4.48E+6	3.18E+6
Rh 106	1.12E+7	9.98E+6	8.40E+6	5.95E+6	5.98E+6	5.30E+6	4.48E+6	3.18E+6
Sb 125	4.17E+5	4.00E+5	3.76E+5	3.31E+5	7.76E+5	7.44E+5	6.99E+5	6.16E+5
Sb 127	8.21E+3	1.39E-1	neg.	neg.	neg.	neg.	neg.	neg.
Te 125m	9.42E+4	9.39E+4	9.04E+4	8.07E+4	1.89E+5	1.82E+5	1.70E+5	1.50E+5
Te 127	2.14E+5	1.40E+5	7.79E+4	2.43E+4	7.15E+3	4.85E+3	2.70E+3	8.44E+2
Te 127m	2.10E+5	1.43E+5	7.95E+4	2.48E+4	7.30E+3	4.95E+3	2.76E+3	8.62E+2
Te 129	2.74E+5	7.79E+4	1.17E+4	2.68E+2	3.85E+0	1.09E+0	1.64E-1	3.76E-3
Te 129m	4.21E+5	1.20E+5	1.79E+4	4.12E+2	5.91E+0	1.68E+0	2.52E-1	5.77E-3
Te 132	3.74E+4	8.64E-2	neg.	neg.	neg.	neg.	neg.	neg.
I 129	8.84E-1	8.86E-1	8.86E-1	8.86E-1	6.26E+0	6.26E+0	6.26E+0	6.26E+0
I 131	1.22E+6	6.35E+3	2.28E+0	neg.	neg.	neg.	neg.	neg.
I 132	3.85E+4	8.90E-2	neg.	neg.	neg.	neg.	neg.	neg.
Xe 133	7.29E+5	2.30E+2	1.21E-3	neg.	neg.	neg.	neg.	neg.
Cs 134	3.53E+6	3.34E+6	3.07E+6	2.59E+6	4.37E+6	4.13E+6	3.80E+6	3.21E+6
Cs 136	2.05E+5	8.13E+3	6.26E+1	3.91E-3	neg.	neg.	neg.	neg.
Cs 137	2.83E+6	2.82E+6	2.80E+6	2.77E+6	1.73E+7	1.73E+7	1.72E+7	1.70E+7
Ba 137m	2.67E+6	2.66E+6	2.65E+6	2.62E+6	1.64E+7	1.63E+7	1.63E+7	1.61E+7
Ba 140	5.19E+6	1.90E+5	1.30E+3	6.41E-2	neg.	neg.	neg.	neg.
La 140	5.97E+6	2.19E+5	1.50E+3	7.37E-2	neg.	neg.	neg.	neg.
Ce 141	1.32E+7	3.61E+6	5.07E+5	1.03E+4	1.31E+2	3.57E+1	5.02E+0	1.01E-1
Ce 144	1.91E+7	1.65E+7	1.32E+7	8.43E+6	7.23E+6	6.23E+6	4.98E+6	3.19E+6
Pr 143	5.44E+6	2.41E+5	2.19E+3	1.90E-1	neg.	neg.	neg.	neg.
Pr 144	1.91E+7	1.65E+7	1.32E+7	8.43E+6	7.23E+6	6.23E+6	4.98E+6	3.19E+6
Nd 147	1.34E+6	3.36E+4	1.05E+2	1.10E-3	neg.	neg.	neg.	neg.
Sm 151	9.31E+3	9.30E+3	9.28E+3	9.25E+3	7.29E+4	7.28E+4	7.26E+4	7.24E+4
Eu 154	2.89E+5	2.85E+5	2.79E+5	2.68E+5	1.05E+6	1.04E+6	1.02E+6	9.75E+5
Eu 156	8.38E+5	5.18E+4	7.76E+2	1.83E-1	neg.	neg.	neg.	neg.
Np 239	5.36E+4	5.26E+2	5.26E+2	5.26E+2	2.35E+5	2.35E+5	2.35E+5	2.35E+5
Pu 238	6.73E+4	6.87E+4	7.02E+4	7.18E+4	3.84E+5	3.84E+5	3.83E+5	3.82E+5
Pu 239	9.28E+3	9.28E+3	9.28E+3	9.28E+3	7.96E+4	7.96E+4	7.96E+4	7.96E+4
Pu 240	1.55E+4	1.55E+4	1.55E+4	1.55E+4	1.15E+5	1.15E+5	1.15E+5	1.15E+5
Pu 241	3.73E+6	3.70E+6	3.65E+6	3.56E+6	1.92E+7	1.91E+7	1.88E+7	1.84E+7
Am 241	6.01E+3	7.00E+3	8.48E+3	1.14E+4	2.82E+5	2.87E+5	2.95E+5	3.09E+5
Cm 242	1.31E+6	1.01E+6	6.86E+5	3.16E+5	1.39E+5	1.08E+5	7.33E+4	3.47E+4
Cm 244	5.88E+4	5.84E+4	5.79E+4	5.68E+4	1.68E+5	1.67E+5	1.65E+5	1.62E+5

[a]Fuel batch 11 is projected discharge during April 1987.
[b]Fuel batches 1-10 were discharged between August 1972 and October 1985.
[c]neg. = less than 10^{-3} Curies.

Table A.7 Decay Heat Released from Spent Fuel Inventory for Various Discharged Fuel Batches (Millstone 1)

	Date, End of Irradiation[a]	Batch Size[a] (Metric Tonnes)	Decay Heat Released by Batch (Kilowatts, Thermal) May 1, 1987	July 1, 1987	April 1, 1988
1	08/31/72	5.13	1.8	1.8	1.8
2	08/31/74	38.13	22.0	21.9	21.5
3	09/11/75	26.40	21.8	21.7	21.2
4	09/30/76	22.73	15.2	15.1	14.8
5	03/10/78	22.73	18.4	18.3	17.7
6	04/27/79	27.13	23.5	23.3	22.4
7	10/03/80	30.79	30.3	29.9	28.2
8	09/11/82	35.19	41.5	40.3	35.9
9	04/12/84	31.53	67.4	63.6	50.9
10	10/01/85[b]	32.63	146.0	132.7	91.8
11	04/01/87[b]	30.61	909.0	537.7	210.5
Total[c] 1-10		272.38	387.9	368.5	306.3
Total[c] 1-11		302.99	1297.0	906.3	516.8

[a]See Table A.3.
[b]Projected dates.
[c]Totals may not equal sum of the entries due to rounding of decimals.

Table A.8 Radionuclide Contributions to Decay Heat for Various Spent Fuel Batches. The Percentage Contributions Depend on the Total Burnup of Each Batch, as well as Decay Time After End of Irradiation (Millstone 1)

Isotope	Spent Fuel Batch Number										
	1	2	3	4	5	6	7	8	9	10	11
	(PERCENT OF TOTAL DECAY HEAT)										
Sr 90	7.48	6.79	6.14	6.61	6.32	6.18	5.85	5.23	3.78	2.34	1.39
Y 90	35.73	32.44	29.33	31.82	30.21	29.52	27.92	24.97	18.06	11.17	4.96
Zr 95	---[a]	---	---	---	---	---	---	---	---	0.01	1.13
Nb 95	---	---	---	---	---	---	---	---	---	0.02	2.33
Rh 106	---	---	---	---	---	0.81	1.89	5.75	13.49	22.53	27.10
Cs 134	0.43	0.98	1.76	2.24	3.74	5.15	7.65	11.63	16.26	16.66	12.53
Cs 137	9.02	8.77	8.43	8.70	8.45	8.28	7.87	6.95	5.16	3.22	1.45
Ba 137m	30.29	29.44	28.30	29.22	28.29	27.80	26.43	23.34	17.34	10.82	4.88
Ce 144	---	---	---	---	---	---	0.06	0.26	0.79	1.73	2.66
Pr 144	---	---	---	---	---	---	0.64	2.93	8.80	19.20	29.42
Eu 154	1.22	2.15	3.03	2.63	3.15	3.32	3.52	3.30	3.03	2.15	1.12
Pu 238	2.14	4.85	7.33	4.66	5.38	5.31	5.33	4.49	3.72	2.37	1.13
Pu 239	2.16	1.54	1.14	1.36	1.16	1.10	0.99	0.88	0.57	0.33	0.14
Pu 240	1.84	1.90	1.79	1.78	1.68	1.61	1.49	1.27	0.90	0.53	0.23
Pu 241	0.19	0.22	0.23	0.24	0.24	0.24	0.24	0.22	0.17	0.11	0.05
Am 241	7.57	7.96	7.34	6.70	5.84	5.12	4.22	2.92	1.61	0.70	0.18
Cm 242	0.01	0.04	0.05	0.02	0.03	0.03	0.03	0.05	0.21	1.21	5.52
Totals[b]	98.08	97.08	94.87	95.98	94.59	94.47	94.13	94.19	93.89	95.10	96.22

[a]Dashes indicate less than 0.01%.

[b]Total percentage of isotopes listed. The balance of the decay heat is distributed among many other less important contributors.

000318

Table A.9 Reactor and Fuel Cycle Parameters for Ginna
(Sources: Refs. 1-4)

Assemblies in core: 121

Licensed thermal power: 1520 MW_{th} (gross)[a]

Thermal power corresponding to maximum dependable capacity: 1499 MW_{th} (gross)

Nominal initial metric tonnes of heave metal (IMTHM) per assembly: 0.375 MT

Average refueling cycle interval (since initial commercial operation): 12.6 months

Average number of assemblies per discharge: 1975-1980: 37
1981-1987: 24

Average IMTHM per discharge: 1975-1908: 15.3 MT
1981-1987: 9.0

Average number of fuel cycles per assembly: 1975-1980: 3.27
1981-1987: 5.04

Average period of irradiation (including down time): 1976-1980: 3.3 years
1981-1987: 5.0 years

Authorized storage pool capacity: 1016

[a]On March 1, 1972 the Atomic Energy Commission authorized an increase in gross thermal power from 1300 to 1520 MW.

000319

Table A.10 Summary of Operational Milestone for Ginna (Source: Ref. 4)

Date of Initial Criticality: November 8, 1969

Date of First Electricity Generation: December 2, 1969

Date of Commercial Operation: July 1, 1970

Lifetime Cumulative Data: (January 1, 1968-March 31, 1986)

Hours, Generator on Line: 107,134.3 hours

Gross Thermal Energy: 149.26×10^6 MWh

Capacity Factor (MDC net): 70.3%

Table A.11 Summary of Spent Fuel Batches in Ginna Storage Basin (With Projections to 1987)

Spent Fuel Batch No.	Date of End of Irradiation	Number of Assemblies	Weight H.M. (MT)	Avg. Burnup (MWD/MT)	Decay[a] Days to 5/1/87 (days)	Cumulative Assemblies in Pool	Cumulative[b] Gross Weight of Spent Fuel in Pool (MT)
1	02/27/71	(37)[c]	14.778	6933	--	0	0
2	04/13/72	(47)[c]	18.772	16695	5832	28	18.4
3	12/31/73	8	3.195	30039	4869	36	23.7
4	03/08/75	29	11.583	38043	4437	65	42.8
5	01/28/76	37	14.778	36958	4111	102	67.1
6	04/14/77	41	16.375	36022	3669	143	94.1
7	03/23/78	41	16.375	27921	3326	184	121.1
8	02/09/79	40	15.976	25451	3003	224	147.4
9	03/28/80	36	14.378	26088	2590	260	171.1
10	04/17/81	28	11.183	27884	2205	288	189.5
11	01/25/82	24	9.586	31054	1891	312	205.3
12	03/25/83	20	7.988	33772	1467	332	218.4
13	03/01/84	23	9.186	37532	1156	355	223.6
14	02/28/85	25	9.985	40533	792	380	250.0
15	03/30/86	24	9.586	42360	397	404	265.8
16[d]	04/01/87	24	9.586	45673	30	428[e]	281.6

[a]Decay days from end of irradiation to 5/1/87.
[b]Gross weight of fuel stored in pool includes heavy metal plus cladding and hardware but not the fuel racks. Each assembly contains approximately 0.4614 tonnes of heavy metal, 0.0620 tonnes of oxygen, 0.1345 tonnes of hardware, totaling 0.6579 tonnes gross.
[c]At the time of the ORIGEN2 calculations some 56 assemblies could not be accounted for using available data.
[d]Projected data.
[e]Authorized capacity is 1016 assemblies.

000321

Table A.12 Comparison of Radioactive Inventories in Reactor Core and Spent Fuel Basin (Ginna)

Isotope	Reactor Core	Spent Fuel Storage Basin[a] 5/1/87	7/1/87	4/1/88
	(Radioactivity, Curies)			
H 3	3.32E+4	9.29E+4	9.20E+4	8.82E+4
C 14	6.42E+1	2.64E+2	2.64E+2	2.64E+2
Co 58	3.57E+5	5.93E+4	3.26E+4	2.21E+3
Co 60	3.20E+5	5.97E+5	5.84E+5	3.29E+5
Kr 85	3.73E+5	9.84E+5	9.74E+5	9.27E+5
Rb 86	6.53E+5	7.22E+3	7.48E+2	2.74E-2
Sr 89	3.55E+7	3.53E+6	1.53E+6	3.50E+4
Sr 90	2.95E+6	1.02E+7	1.01E+7	9.95E+6
Y 90	3.15E+6	1.02E+7	1.01E+7	9.95E+6
Y 91	4.37E+7	5.11E+6	2.48E+6	9.34E+4
Zr 95	6.41E+7	8.64E+6	4.46E+6	2.27E+5
Nb 95	6.34E+7	1.12E+7	7.51E+6	4.93E+5
Mo 99	6.83E+7	7.03E+3	1.48E-3	neg.[b]
Tc 99m	5.89E+7	6.77E+3	1.42E-3	neg.[b]
Ru 103	5.85E+7	7.86E+6	2.88E+6	2.09E+4
Ru 106	1.95E+7	1.09E+7	9.71E+6	5.78E+6
Rh 106	2.15E+7	1.09E+7	9.71E+6	5.78E+6
Sb 125	6.04E+5	7.11E+5	6.82E+5	5.65E+5
Sb 127	4.12E+6	4.33E+3	7.35E-2	neg.[b]
Te 125m	1.27E+5	1.70E+5	1.65E+5	1.37E+5
Te 127	4.05E+6	1.19E+5	7.79E+4	1.36E+4
Te 127m	5.19E+5	1.17E+5	7.95E+4	1.38E+4
Te 129	1.21E+7	1.38E+5	3.93E+4	1.35E+2
Te 129m	1.80E+6	2.12E+5	6.03E+4	2.07E+2
Te 132	5.33E+7	1.83E+4	4.23E-2	neg.[b]
I 129	1.27E+0	5.32E+0	5.32E+0	5.32E+0
I 131	3.76E+7	6.00E+5	3.12E+5	neg.[b]
I 132	5.42E+7	1.89E+4	4.36E-2	neg.[b]
Xe 133	7.64E+7	3.52E+5	1.11E+2	neg.[b]
Cs 134	5.82E+6	6.35E+6	6.00E+6	4.66E+6
Cs 136	1.87E+6	1.26E+5	4.99E+3	2.40E-3
Cs 137	4.21E+6	1.48E+7	1.47E+7	1.44E+7
Ba 137m	4.00E+6	1.40E+7	1.39E+7	1.37E+7
Ba 140	6.55E+7	2.47E+6	9.07E+4	3.05E-2
La 140	6.74E+7	2.85E+6	1.04E+5	3.51E-2
Ce 141	6.28E+7	6.34E+6	1.72E+6	4.91E+5
Ce 144	4.24E+7	1.38E+7	1.19E+7	6.09E+6
Pr 143	5.71E+7	2.54E+6	1.12E+5	8.86E-2
Pr 144	4.27E+7	1.38E+7	1.19E+7	6.09E+6
Nd 147	2.48E+7	7.42E+5	1.62E+4	neg.[b]
Sm 151	1.42E+4	5.14E+4	5.13E+4	5.10E+4
Eu 154	4.09E+5	1.09E+6	1.07E+6	1.01E+6
Eu 156	7.22E+6	7.58E+5	4.68E+4	1.66E-1
Np 239	7.81E+8	3.02E+4	3.26E+3	3.26E+3
Pu 238	1.01E+5	4.46E+5	4.46E+5	4.46E+5
Pu 239	1.35E+4	5.25E+4	5.25E+4	5.25E+4
Pu 240	2.02E+4	8.60E+4	8.60E+4	8.61E+4
Pu 241	4.85E+6	1.52E+7	1.51E+7	1.46E+7
Am 241	4.99E+3	2.10E+5	2.14E+5	2.32E+5
Cm 242	1.91E+6	9.33E+5	7.20E+5	2.25E+5
Cm 244	1.25E+5	3.59E+5	3.56E+5	3.46E+5

[a]Spent fuel pool inventory includes discharges from 13 refuelings covering the period from April 1983 through the projected refueling of April 1987.

[b]neg. = less than 10^{-3} Curies.

Table A.13 Comparison of Radioactive Inventories of Most Recently Discharged Fuel Batch (Batch 16) with Longer Aged Discharged Batches (Batches 2-15) (Ginna)

Isotope	Spent Fuel Batch 16[a]				Spent Fuel Batch 2-15[b]			
	5/1/87	7/1/87	10/1/87	4/1/88	5/1/87	7/1/87	10/1/87	4/1/88
	(Radioactivity, Curies)							
H 3	9.89E+3	9.80E+3	9.66E+3	9.39E+3	8.29E+4	8.22E+4	8.10E+4	7.88E+4
C 14	2.20E+1	2.20E+1	2.20E+1	2.20E+1	2.42E+2	2.42E+2	2.42E+2	2.42E+2
Co 58	5.77E+4	3.18E+4	1.29E+4	2.15E+3	1.60E+5	8.78E+2	3.57E+2	5.94E+1
Co 60	9.92E+4	9.70E+4	9.39E+4	8.79E+4	4.98E+5	4.87E+5	4.71E+5	4.41E+5
Kr 85	1.07E+5	1.05E+5	1.04E+5	1.00E+5	8.78E+5	8.68E+5	8.54E+5	8.27E+5
Rb 86	7.22E+3	7.48E+2	2.45E+1	2.73E-2	neg.	neg.	neg.	neg.
Sr 89	3.50E+6	1.52E+6	4.29E+5	3.48E+6	2.39E+4	1.04E+4	2.93E+3	2.38E+2
Sr 90	8.56E+5	8.53E+5	8.48E+5	8.38E+5	9.32E+6	9.28E+6	9.23E+6	9.12E+6
Y 90	8.57E+5	8.53E+5	8.48E+5	8.38E+5	9.32E+6	9.28E+6	9.23E+6	9.12E+6
Y 91	5.04E+6	2.45E+6	8.23E+5	9.41E+4	6.86E+4	3.33E+4	1.12E+4	1.28E+3
Zr 95	8.47E+6	4.37E+6	1.62E+6	2.23E+5	1.64E+5	8.48E+4	3.13E+4	5.60E+4
Nb 95	1.09E+7	7.33E+6	3.19E+6	4.83E+5	3.68E+5	1.89E+5	6.96E+4	9.57E+3
Mo 99	7.03E+3	1.48E-3	neg.[c]	neg.	neg.	neg.	neg.	neg.
Tc 99m	6.77E+3	1.42E-3	neg.	neg.	neg.	neg.	neg.	neg.
Ru 103	7.86E+6	2.68E+6	5.28E+5	2.09E+4	1.18E+4	4.02E+3	7.93E+2	3.14E+1
Ru 106	5.82E+6	5.19E+6	4.37E+6	3.09E+6	5.06E+6	4.51E+6	3.80E+6	2.69E+6
Rh 106	5.82E+6	5.19E+6	4.37E+5	3.09E+6	5.06E+6	4.51E+6	3.80E+6	2.69E+6
Sb 125	1.84E+5	1.76E+5	1.65E+5	1.46E+5	5.28E+5	5.06E+5	4.75E+5	4.19E+5
Sb 127	4.33E+3	7.35E-2	neg.	neg.	neg.	neg.	neg.	neg.
Te 125m	4.13E+4	4.12E+4	3.97E+4	3.55E+4	1.78E+5	1.24E+5	1.16E+5	1.02E+4
Te 127	1.08E+5	7.05E+4	3.93E+4	1.23E+4	1.09E+4	7.42E+3	4.14E+3	1.29E+4
Te 127m	1.06E+5	7.19E+4	4.01E+4	1.25E+4	1.12E+4	7.58E+3	4.22E+3	1.32E+3
Te 129	1.38E+5	3.93E+4	5.88E+3	1.35E+2	6.98E+1	1.98E+1	2.97E+0	6.82E-2
Te 129m	2.12E+5	6.03E+4	9.04E+3	2.07E+2	1.07E+2	3.05E+1	4.57E+0	1.05E-1
Te 132	1.83E+4	4.23E-2	neg.	neg.	neg.	neg.	neg.	neg.
I 129	4.22E-1	4.23E-1	4.23E-1	4.23E-1	4.89E+0	4.89E+0	4.89E+0	4.89E+0
I 131	6.00E+5	3.12E+3	1.12E+0	neg.	neg.	neg.	neg.	neg.
I 132	1.89E+4	4.36E-2	neg.	neg.	neg.	neg.	neg.	neg.
Xe 133	3.52E+5	1.11E+2	neg.	neg.	neg.	neg.	neg.	neg.
Cs 134	2.26E+6	2.13E+6	1.96E+6	1.66E+6	4.09E+6	3.87E+6	3.55E+6	3.01E+6
Cs 136	1.26E+5	4.99E+3	3.84E+1	2.40E-3	neg.	neg.	neg.	neg.
Cs 137	1.34E+6	1.34E+6	1.33E+6	1.31E+6	1.34E+7	1.34E+7	1.33E+7	1.31E+7
Ba 137m	1.27E+6	1.26E+6	1.26E+6	1.24E+6	1.27E+7	1.26E+7	1.26E+7	1.24E+7
Ba 140	2.47E+6	9.07E+4	6.19E+2	3.05E-2	neg.	neg.	neg.	neg.
La 140	2.85E+6	1.04E+5	7.13E+2	3.51E-2	neg.	neg.	neg.	neg.
Ce 141	6.34E+6	1.73E+6	2.43E+5	4.91E+3	2.53E+3	6.89E+2	9.69E+1	1.96E+0
Ce 144	8.25E+6	7.11E+6	5.68E+6	3.64E+6	5.58E+6	4.81E+6	3.84E+6	2.46E+6
Pr 143	2.54E+6	1.12E+5	1.02E+3	8.86E-2	neg.	neg.	neg.	neg.
Pr 144	8.25E+6	7.11E+6	5.68E+6	3.64E+6	5.58E+6	4.81E+6	3.84E+6	2.46E+6
Nd 147	7.42E+5	1.62E+4	5.08E+1	neg.	neg.	neg.	neg.	neg.
Sm 151	3.47E+3	3.47E+3	3.46E+3	3.45E+3	4.79E+4	4.79E+4	4.78E+4	4.76E+4
Eu 154	1.67E+5	1.65E+5	1.61E+5	1.55E+5	9.19E+5	9.06E+5	8.88E+5	8.53E+5
Eu 156	7.58E+5	4.68E+4	7.02E+2	1.66E-1	neg.	neg.	neg.	neg.
Np 239	2.74E+4	4.59E+2	4.59E+2	4.59E+2	2.80E+3	2.80E+3	2.80E+3	2.80E+3
Pu 238	4.87E+4	4.95E+4	5.04E+4	5.13E+4	3.97E+5	3.97E+5	3.96E+5	3.95E+5
Pu 239	3.05E+3	3.05E+3	3.05E+3	3.05E+3	4.95E+4	4.95E+4	4.95E+4	4.95E+4
Pu 240	6.01E+3	6.01E+3	6.01E+3	6.02E+3	8.00E+4	8.00E+4	8.00E+4	8.00E+4
Pu 241	1.58E+6	1.57E+6	1.55E+6	1.51E+6	1.37E+7	1.35E+7	1.34E+7	1.31E+7
Am 241	2.05E+3	2.47E+3	3.10E+3	4.33E+3	2.08E+5	2.12E+5	2.17E+5	2.28E+5
Cm 242	7.57E+5	5.85E+5	3.96E+5	1.82E+5	1.75E+5	1.36E+5	9.21E+4	4.30E+4
Cm 244	8.06E+4	8.00E+4	7.93E+4	7.78E+4	2.78E+5	2.76E+5	2.74E+5	2.68E+5

[a]Fuel batch 16 is projected discharge during April 1987.
[b]Fuel batches 2-15 were discharged between April 1972 and April 1986.
[c]neg. = less than 10^{-3} Curies.

Table A.14 Decay Heat Released from Spent Fuel Inventory for Various Discharged Fuel Batches (Ginna)

	Date, End of Irradiation[a]	Batch Size[a] (Metric Tonnes)	Decay Heat Released by Batch (Kilowatts, Thermal) May 1, 1987	July 1, 1987	April 1, 1988
2	04/13/72	18.772	8.5	8.5	8.4
3	12/31/73	3.195	2.9	2.9	2.8
4	03/08/75	11.583	14.3	14.2	13.9
5	01/28/76	14.778	18.1	18.0	17.6
6	04/14/77	16.375	20.5	20.4	19.8
7	03/23/78	16.375	15.8	15.7	15.1
8	02/09/79	15.976	14.7	14.5	14.0
9	03/28/80	14.378	14.7	14.5	13.7
10	04/17/81	11.183	13.7	13.4	12.4
11	01/25/82	9.586	15.0	14.6	13.2
12	03/25/83	7.988	17.2	16.5	14.2
13	03/01/84	9.186	28.6	27.1	22.0
14	02/28/85	9.985	50.9	47.2	35.3
15	03/30/86	9.586	96.1	85.8	56.5
16	04/01/87[b]	9.586	437.2	260.4	107.7
Total[c] 2-15			331.0	313.3	259.0
Total[c] 2-16			768.3	573.7	366.8

[a]See Table A.11.
[b]Projected dates.
[c]Totals may not equal sum of entries due to rounding of decimals.

Table A.15 Radionuclide Contributions to Decay Heat for Various Spent Fuel Batches. The Percentage Contributions Depend on the Total Burnup of Each Batch, as well as Decay Time After End of Irradiation (Ginna)

Isotope	Spent Fuel Batch Number														
	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
	(PERCENT OF TOTAL DECAY HEAT)														
Sr 90	7.32	6.26	5.56	5.61	5.61	6.23	6.32	6.07	5.61	5.01	4.23	3.43	2.61	1.60	0.90
Y 90	34.95	29.89	26.57	26.82	26.79	29.77	30.20	29.01	26.79	23.95	20.19	16.37	12.01	7.64	4.31
Zr 95	---[a]	---	---	---	---	---	---	---	---	---	---	---	---	0.04	1.01
Nb 95	---	---	---	---	---	---	---	---	---	---	---	---	---	0.08	2.08
Rh 106	---	0.02	0.05	0.08	0.19	0.41	0.76	1.46	2.86	4.65	8.19	12.24	18.16	24.96	27.54
Cs 134	0.33	1.12	1.77	2.26	3.29	4.01	4.95	6.72	9.38	12.21	15.67	18.46	20.18	19.07	15.65
Cs 137	8.89	8.42	7.93	7.94	7.89	8.23	8.19	7.89	7.38	6.75	5.81	4.83	3.63	2.33	1.35
Ba 137m	29.85	28.26	26.64	26.67	26.48	27.65	27.50	26.51	24.79	22.68	19.51	16.22	12.18	7.84	4.52
Ce 144	---	---	---	---	---	0.01	0.02	0.05	0.11	0.19	0.39	0.63	1.07	1.78	2.24
Pr 144	---	---	---	0.01	0.03	0.09	0.23	0.52	1.21	2.13	4.31	6.99	11.84	19.75	24.80
Eu 154	1.33	2.73	3.37	3.48	3.66	3.21	3.05	3.20	3.41	3.64	3.60	3.41	2.88	2.02	1.29
Pu 238	2.90	6.98	9.46	9.12	8.67	5.77	4.76	4.71	4.83	5.18	5.06	4.82	3.95	2.58	1.58
Pu 239	1.84	1.07	0.81	0.82	0.80	1.01	1.05	0.97	0.84	0.69	0.54	0.41	0.28	0.17	0.09
Pu 240	1.80	1.69	1.49	1.49	1.45	1.53	1.47	1.38	1.27	1.13	0.94	0.74	0.53	0.32	0.17
Pu 241	0.17	0.21	0.20	0.20	0.21	0.22	0.22	0.22	0.21	0.20	0.18	0.15	0.11	0.08	0.04
Am 241	7.88	7.68	6.60	6.20	5.56	5.19	4.67	4.07	3.31	2.69	1.97	1.39	0.82	0.37	0.13
Cm 242	0.01	0.03	0.03	0.03	0.03	0.02	0.02	0.02	0.02	0.03	0.07	0.21	0.70	2.20	6.22
Cm244	0.25	2.74	6.45	5.94	5.63	2.18	1.52	1.64	2.03	2.87	3.47	4.25	4.35	3.32	2.52
Totals[b]	97.52	97.10	96.93	96.67	96.29	95.53	94.93	94.44	94.05	94.00	94.13	94.55	95.20	96.15	96.44

[a]Dashes indicate less than 0.01%.
[b]Total percentage of isotopes listed. The balance of the decay heat is distributed among many other less important contributors.

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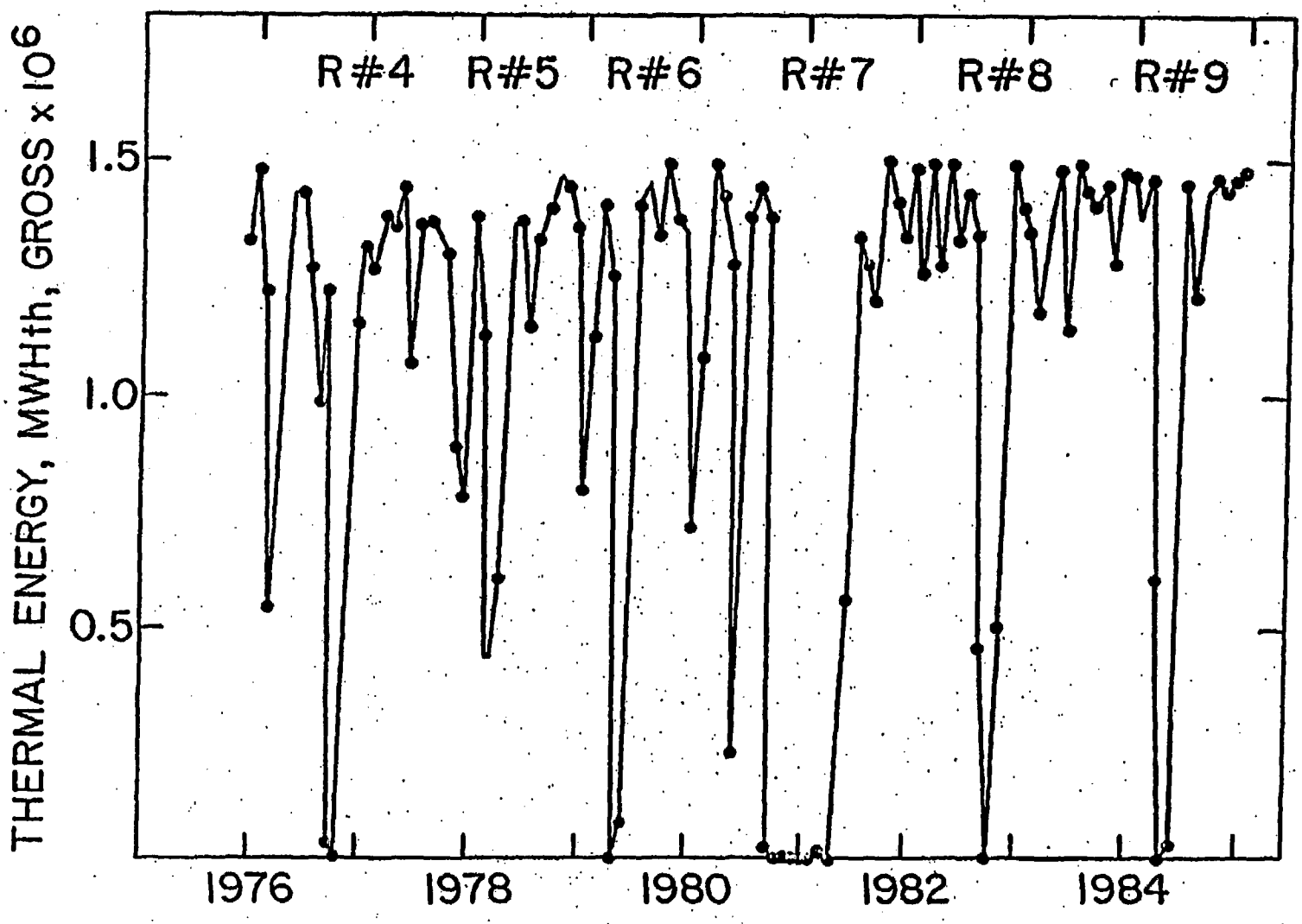

Figure A.1 Millstone 1: Operating history 1976-1984 (Source: Ref. 4).

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APPENDIX B

IMPACT OF REVISED REACTION RATE EQUATION ON THE LIKELIHOOD OF ZIRCONIUM FIRES IN A DRAINED SPENT FUEL POOL

The SNL investigation[1] of the potential for cladding oxidation during loss of fuel pool inventory accidents has been controversial due to many unique features of the postulated "beyond design basis accident." The purpose of the BNL investigation has been two-fold:

1. Provide an independent assessment of several important areas of the phenomenological treatment of the SFUEL code.[1]

2. Provide an estimate of the likelihood and consequences of the postulated accidents so that the risk can be compared to the risk of severe reactor accidents evaluated in typical PRAs.

The purpose of this appendix is to re-evaluate the oxidation rate equation used in the SFUEL code and to perform a sensitivity study to demonstrate the influence of the reaction rate on the results of the SFUEL analysis.

The oxidation rate equation is also a key factor which affects the possible propagation of Zircaloy fires to low power (i.e., older) spent fuel bundles. Based on the information available to date it appears impossible to justify any major changes to the Sandia equation; in particular, the curve from the work of White at temperatures above 1150°C appears to be all we have. However, this was obtained on unalloyed Zr, not Zircaloy, and the higher rates for Zircaloy-4 over those for unalloyed Zr observed by Pawel and Campbell in steam may also exist in air. For temperatures from 800-1150°C, the new German data fit in well with the previous data. However, if the exposure is for periods greater than 30 minutes, this curve may not be conservative, as shown in the new German data plotted in Figure B.3.

After an extensive review of the zirconium/Zircaloy reaction rate data and recent unpublished data, BNL has concluded that the reaction rate in Equation (1) is representative of the existing data.

$$w^2/t = 3.09 \times 10^8 \exp(-56600/RT) \qquad (1)$$

where: w is the oxygen consumption (mg/cm^2)
t is time (sec)
T is the clad temperature (K)
R is the gas constant (1.987 cal/K)

This equation is equivalent to that suggested by Benjamin et al.[1] except that it provides a smooth transition to the self-sustaining oxidation regime (above 800°C) and does not put undue emphasis on the threshold effect of a shift in oxidation rate due to metallic phase change.

The reaction rate has been varied by a factor of four based on the data scatter in the temperature range of 800 to 900°C (where self-sustaining oxidation is initiated). Only a slight change (±50°C) in the initiation temperature occurs for this broad range of uncertainty in the oxidation rate. This translates into an uncertainty of ±25% in the critical decay power. This

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insensitivity to the oxidation rate equation basically confirms the SNL analysis[1] for zirconium fire initiation in a dry spent fuel pool.

As Benjamin et al.[1] pointed out, the most sensitive parameters for clad fire initiation are the decay heat level and the fuel rack geometry (related to natural circulation flow resistance). Thus, for BWRs with low power density and relatively open fuel storage racks, the critical cooling time (to ensure that air cooling will keep the fuel rods below 800°C) is about 1 to 5 months. Whereas PWRs with higher power density and tighter storage racks require 2 months to 2 years (the longer time is required for the new high density storage racks).

Note that even temperatures as low as 650°C can be expected to cause clad failure and release of some fission products if the temperatures are sustained over a long period (several hours). However, below 800°C the energy from oxidation is insufficient to significantly increase the fuel rod temperature.

The results indicate that the SNL code (SFUEL) and the clad oxidation rate equation used therein accurately represents the potential for self-sustaining oxidation in a drained fuel pool. The largest uncertainty appears to be due to uncertainties in natural convection flows in the transition flow regime. Changes in the storage rack configuration result in large changes in the calculated flow rate and correspondingly large changes in the "critical power level" (above which self-sustaining oxidation is predicted to occur).

Based on the review of the cladding oxidation rate model and the sensitivity study, it is concluded that the conditional probability of self-sustaining clad oxidation and resultant fission product release, given a loss of pool integrity event, is about 10% to 40% for BWRs and 16% to 100% for PWRs, depending on the storage rack configuration.

In terms of power level, our sensitivity studies indicate that the critical power level (above which self-sustaining oxidation will occur) varies from about 50 kW/MTU (for cylindrical racks with large openings) to 6 kW/MTU for the new high density PWR fuel storage racks.

In order to ensure that self-sustaining oxidation in the current high density racks will not occur, it is recommended that spent fuel not be stored in high density racks until it has been stored for 2 or more years in the old style cylindrical racks with adequate coolant openings (3 or more inch diameter holes).

It is also recommended that a test program be initiated to confirm the capability of natural air convection cooling capability for high density storage racks. Such tests could be performed with old low power spent fuel (2 to 4 kw/MTU) and minimal instrumentation (such as thermocouples placed near the top of the fuel bundle).

Two Sandia reports[1,2] deal with the question of rapid zirconium oxidation in a spent fuel pool following loss of water. Both the computer modeling and the experimental simulation, as described in these reports, suggested that in certain fuel racking configurations (a) a self-sustaining zirconium-air oxidation reaction can be initiated, and (b) this self-sustaining reaction can propagate from one region of a pool to another. There are large uncertainties

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associated with the phenomenology of Zircaloy oxidation and its propagation in spent fuel assemblies. This review addresses some of these uncertainties and their effects on the initiation and propagation of a self-sustaining Zircaloy-air oxidation reaction.

1. The propagation rates of rapid Zircaloy clad oxidation in air from the hottest section of the pool (after a loss of water incident) to adjacent sections were estimated (in Ref. 2) under the conditions that the spent fuel in the hottest section of the pool was generating 30 kw/MTU in a room maintained at constant temperature. As pointed out by Han,[3] this estimate should be re-calculated under inadequate room ventilation conditions, to simulate properly the conditions at many licensed facilities. Similarly, additional calculations should be performed in which the hot spent fuel decay power is varied from 20 to 90 kw/MTU for both the adequate and inadequate room ventilation conditions. These studies would determine how sensitive the oxidation propagation is to the decay power of the spent fuel stored adjacent to hot fuel, assuming the input oxidation rate data are known with sufficient accuracy.

2. The above assumes the Zircaloy-air reaction rate equation used in the Sandia work is sufficiently accurate. There are a number of uncertainties associated with this equation. We discuss each of these uncertainties in turn.

   A. Experimental Data: A literature search[4-147] has revealed that there is a great deal of data for zirconium oxidation; most of it, however, is concerned with oxidation in steam or oxygen. The data for zirconium (Zircaloy)-air oxidation presented in Refs. 1 and 2 appear to be the best available. These are shown in Figure B.1. The authors (of the SNL reports) fit the data with three separate Arrhenius plots over the temperature range 500-1500°C; one break occurs at the α-β transformation temperature for zirconium, the other at the temperature at which the oxide undergoes a monoclinic-tetragonal transformation. (N.B. two of the sets of data are for zirconium, the other for Zircaloy-4). These assumptions are reasonable. It should be noted, however, that there is no a priori reason to expect that the data would be fit by an Arrhenius expression, particularly above the α-β transformation temperature where a number of different processes are occurring simultaneously (discussed further below); therefore the use of the Arrhenius expression should be viewed in this case only as a computational tool. It is difficult to assess the validity of the data employed. What are really required are new experiments to determine the oxidation rate of Zircaloy in air over the temperature range of interest, for both isothermal and non-isothermal conditions.

   B. Kinetics: The questions was raised[2] as to whether the assumption of parabolic kinetics was valid. Data were presented (from Refs. 86 and 126) which show examples of linear as well as cubic kinetics. However, they all apply at temperatures below the α-β transformation temperature. Since almost all rapid oxidation occurs above the α-β transformation temperature, where the oxidation rate is controlled by

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one or more diffusion processes, the assumption of parabolic kinetics appears to be reasonable.

C. Zirconium vs. Zircaloy: It is assumed in the Sandia work that the oxidation rates of zirconium and Zircaloy are essentially the same. Recent work by Pawel and Campbell[136] has shown that this is not the case. Oxidation in steam of both pure zirconium and Zircaloy-4 was studied in the temperature range of rapid oxidation (1000°C-1500°C). It was found that at all temperatures the oxidation rate of Zircaloy-4 was higher than that of zirconium; the ratio of the two rates is approximately 3 at 1000°C and decreases with increasing temperature to a value of approximately 1.5 at 1500°C (cf. Figure B.2). The higher oxidation rate of Zircaloy-4 is attributed to increased oxygen diffusivity in the oxide phase; a lower activation energy was observed, implying that some mechanistic differences exist. Analogous results are expected to apply for oxidation in air.

D. Oxidation Model: The oxidation in steam of both zirconium and Zircaloy-4 (in the temperature range 1000-1500°C) is a multi-phase layer process.[136] Not only is an oxide layer formed, but also (beneath it) a layer of oxygen-stabilized α-phase (zirconium or Zircaloy). The multi-phase model is only significant above the α-β transformation temperature (approximately 900°C), but this is exactly where rapid oxidation occurs. The parabolic rate constants for oxide layer growth, α-layer growth, and oxygen consumption were determined in Ref. 136 from experimental data and computer modeling. The rate of oxygen consumption is significantly higher at all temperatures than the rate of oxide formation for both zirconium and Zircaloy-4. For zirconium the ration of oxygen consumption rate to oxidation rate is approximately 4 at 1000°C and increases with increasing temperature to a value of approximately 5.4 at 1500°C; for Zircaloy-4 the corresponding values are approximately 3.0 and 4.5 at 1000°C and 1500°C, respectively (cf. Figure B.2). Although these results were obtained for oxidation in steam, analogous results are again expected for oxidation in air.

E. Effect of Nitrogen: Before discussing the reaction of zirconium with air, let us consider the reaction with nitrogen alone.[148-151] The rate of reaction of nitrogen with zirconium is much less than the corresponding reaction rate with oxygen; weight gain data[151] after one hour (800°C<T<1200°C) indicate that zirconium reacts with nitrogen about 20 times slower than with oxygen. The overall process is very similar to oxidation in view of the high solubility of nitrogen in zirconium, and involves a large amount of dissolution along with film formation. In the case of nitriding in the α-region, a two phase diffusion process describes the behavior whereas β-phase nitriding involves three phases (nitrogen, like oxygen, stabilizes the α-phase, leading to a wide range of α between the nitride and the β-matrix). The reaction product is zirconium nitride (ZrN); the reaction is exothermic, releasing approximately 82 kcal/mole. (The energy released in forming the oxide is approximately 262 kcal/mole.) The thickness of the zirconium nitride layer has been found[149] to be much smaller than that of the dissolution zone (in the temperature range 750°C-1000°C) which indicates that the rate constant for film

formation is considerably smaller than the rate constant for nitrogen dissolution. In fact, at 1000°C, 84% of the total nitrogen uptake was due to dissolution in the metal.

The role of nitrogen in the high temperature reaction of zirconium with air has been investigated.[151] The reaction process is multi-phase in nature. Adjacent to the β-phase of the zirconium is a layer of α-phase (stabilized by both oxygen and nitrogen) and a surface layer of ZrO_2. In general, a certain amount of nitride (ZrN) is formed. For temperatures up to approximately 1050°C the nitride is found as a layer between the stabilized α-phase and the oxide layer; above 1050°C the nitride occurs as discrete particles dispersed in the oxide.

It is doubtful whether any appreciable amount of nitride is formed in the problem currently being considered. At the lower temperatures (during heat up) the reaction rate is very slow. Once rapid oxidation is initiated (approximately 900°C) the self-sustaining reaction proceeds very quickly, and there may not be sufficient time for ZrN to be formed. Any nitride that does form, however, will contribute to the chemical energy release for the self-sustaining reaction.

The reaction rate of zirconium is higher with air than with oxygen alone. The explanation advanced is that nitrogen dissolves in ZrO_2. By replacing oxygen ions in the oxide structure, the higher valency nitrogen can increase the anion vacancy concentration, thus permitting a higher rate of diffusion of oxygen through the anion-deficient zirconia.

There are a number of uncertainties associated with the Zircaloy-air reaction equation. These are particularly important above 900°C where rapid oxidation occurs. The most significant appear to be (i) the difference in the oxidation rates of zirconium and Zircaloy, and (ii) the multi-phase nature of the oxidation process itself at these temperatures. The results given above in Section C and D (i.e., for zirconium vs. Zircaloy-4, and oxygen consumption rate vs. oxidation rate, respectively) apply to oxidation in steam only. Analogous results are expected for oxidation in air, i.e., it is expected that the oxidation rate in Zircaloy will be greater than that in zirconium, and the rate of oxygen consumption will be greater than the rate of oxide formation in both materials. The relative magnitude of these effects cannot be deduced from the steam oxidation data. What are required are new experiments and computer modeling (similar to those carried out by Pawel and Campbell[136] for oxidation in steam) for the high temperature reaction of zirconium and Zircaloy with air. In lieu of these, we suggest that additional calculations be performed for two other zirconium-air reaction correlations which will serve as bounds for those presented in Figure B.1. (a) The high temperature correlation for zirconium (above the phase change of ZrO_2) should be multiplied by a factor m_1 to account for the higher reaction rate in Zircaloy. (b) The correlations above the α-β transformation temperature should be divided by a factor m_2 to account for

the difference in oxygen consumption rate and rate of oxide formation. Values of m_1 and m_2 as large as five should be considered.

F. New Data: The Pawel and Campbell data are compared with the Sandia curve in Figure B.3. It is immediately apparent that the Pawel and Campbell[152] parabolic rate constants are considerably lower than the curve used by Sandia. While the rate controlling step in the high temperature oxidation of zirconium or Zircaloys is the diffusion of oxygen through the oxide and/or though the solid solution of oxygen in Zircaloy that underlies it in both steam and air oxidation, there is a significant decrease in the oxidation rate observed in a steam environment due to an effect of the hydrogen produced during this oxidation on these diffusion constants. This effect has been observed by several workers, but it is not sufficiently quantified to permit the use of high temperature steam data (such as those of Prater and Courtright[153] and those of Pawel and Campbell[152]) to estimate oxidation rates under the fuel pool accident scenario. This leaves only the White[124] data in the high temperature range.

Some new unpublished data from Leistikow show roughly a parabolic corrosion rate behavior (slope of 1/2 on the log log plot) for the first 30-60 minutes in both air and steam. They also show that the difference between the air and the steam rates increases with temperature. After 30-60 minutes, however, the rate at all but the highest temperature increased dramatically, especially in air. This may be due either to difficulty in controlling the temperature of the highly exothermic zirconium/air oxidation, or to some "breakaway" type phenomenon in the surface oxides exposing the bare metal underneath.

At lower temperatures, zirconium and Zircaloy are known to oxidize according to the cubic law, which would mean a slope of 1/3 on a log log plot. The high temperature data used by Benjamin et al.[1] were all approximated using parabolic growth, which is more typical of diffusion controlled phenomenon such as are believed to occur at high temperatures. The new German data show a slope somewhere between 1/3 and 1/2 for the first 30 minutes or so. In an attempt to compare these data with the Sandia curve, a parabolic rate constant was calculated for Leistikow's data and compared with the Sandia curves and the Pawel and Campbell data in steam in Figure B.3. It is apparent that the German steam data and those of Pawel and Campbell for Zircaloy-4 in steam are consistent in the temperature range in which they overlap. The new German air data are consistent with some of their own work (at short exposure times) published some years earlier.[125] The new German data can be approximated by the rate equation:

$$\frac{(w)^2}{t} = 3.09 \times 10^8 \exp\left(-\frac{56,600}{RT}\right) \qquad (2)$$

where w is in mg O_2 reacted per square cm, t is in seconds, and T is in °K.

The instantaneous rate, dw/dt at time t and temperature T is given by

$$\frac{dw}{dt} = \frac{1}{2w} 3.09 \times 10^8 \exp\left(-\frac{56,600}{RT}\right) \qquad (3)$$

The Sandia[1] curve shows an abrupt increase in oxidation rate at $10^4/T = 7$, which was attributed to the mono-tetragonal phase change of ZrO_2. As can be seen in Figure B.3, the Pawel and Campbell data do not show such an abrupt change at this temperature; however, these data were obtained in steam. The recent results of Prater and Courtright[153] (which were presented at the 1985 Symposium on Zirconium) show that for reactions in steam they find a similar jump at temperatures as high as 1500°C (1/T is 5.5×10^{-4}). This may be due to effects of the hydrogen produced by steam reaction on the oxide structure on the Zircaloy. Unfortunately, Prater and Courtright plotted their data in terms of thickness of the ZrO_2 film, and thus these could not readily be transferred to Figure B.3 which is in wt. of O_2 reacted. Since a considerable amount of the oxygen that reacts either from air or steam exists in high concentration solid solutions in the Zircaloy, and since the heat generated by this reaction is the main concern, it is important that the total oxygen consumption be considered rather than just the thickness of the layer. It is anticipated that the free energy of formation per gram atom of oxygen reacted would be approximately the same for the zirconium oxygen solid solution as for ZrO_2 at these high temperatures.

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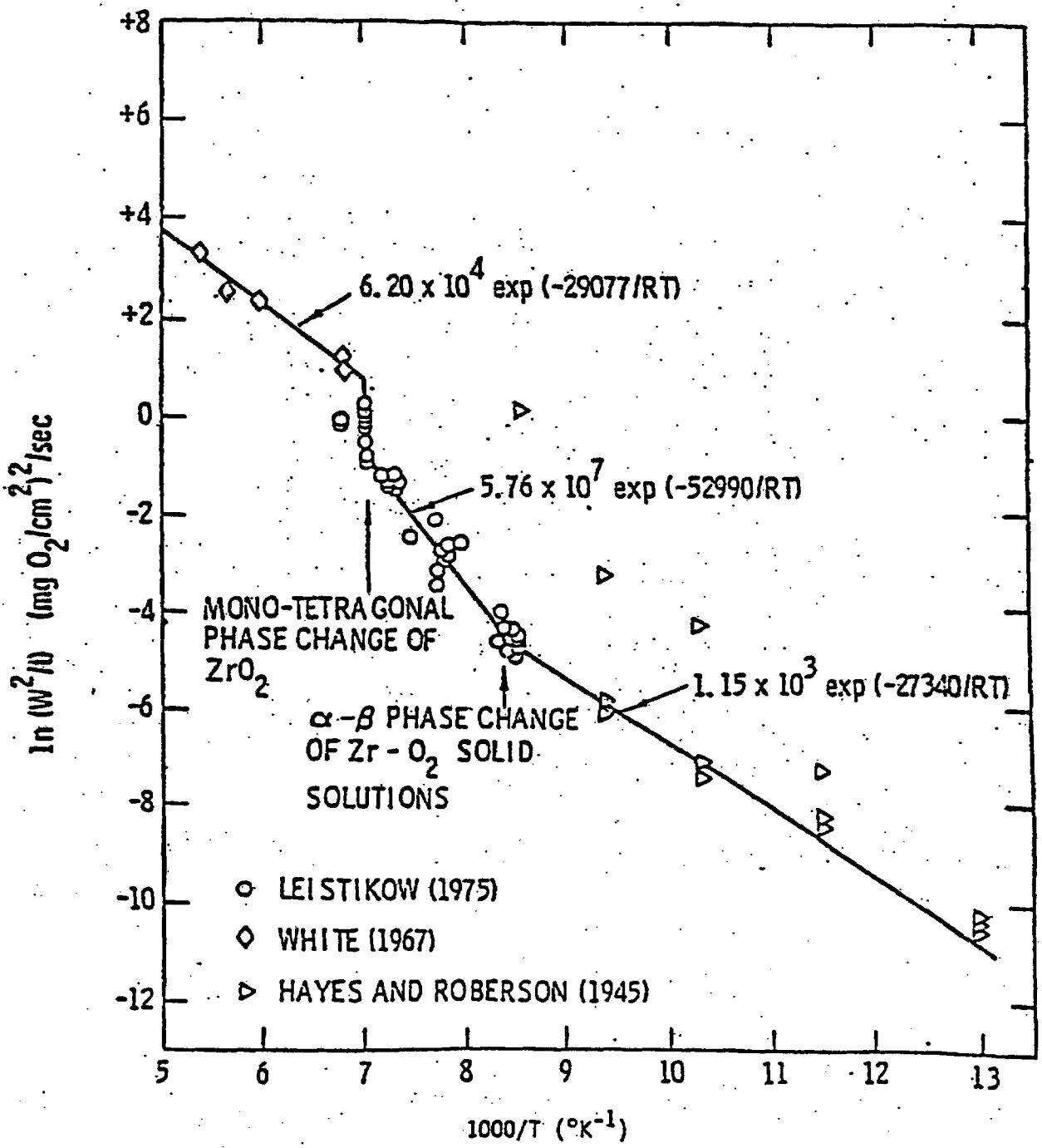

Figure B.1 Correlations for zirconium and Zr-4 oxidation in air (from Ref. 1).

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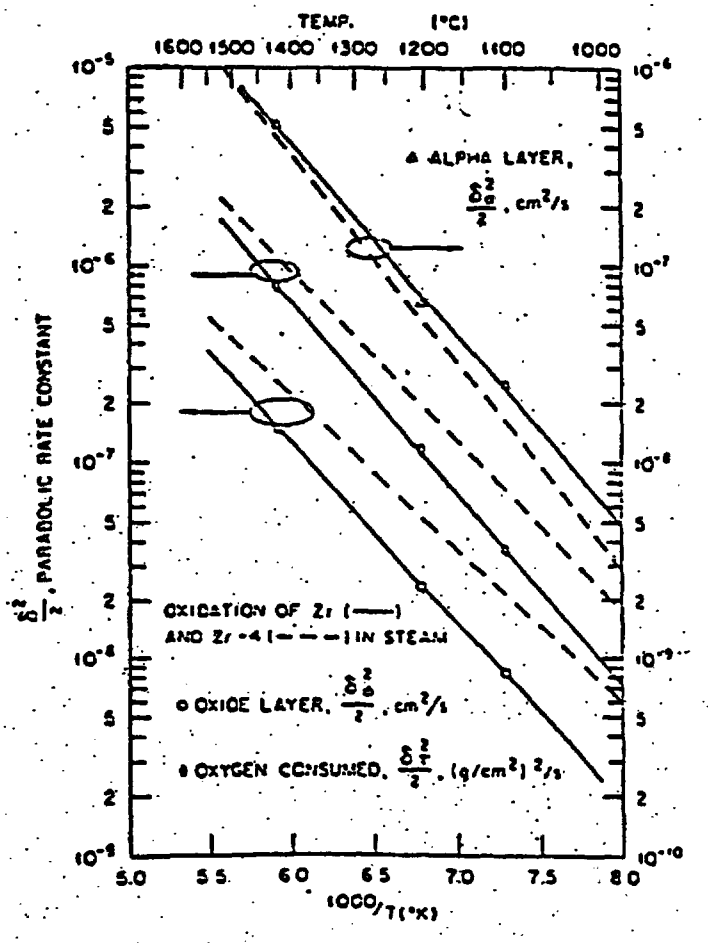

Figure B.2 Parabolic rate constants for oxide layer growth, α-layer growth, and oxygen consumption for the reaction of zirconium (solid lines) and Zircaloy-4 (dashed lines) with steam. The rate constants for oxygen consumed (weight gain) were determined from modeling analyses (from Ref. 136).

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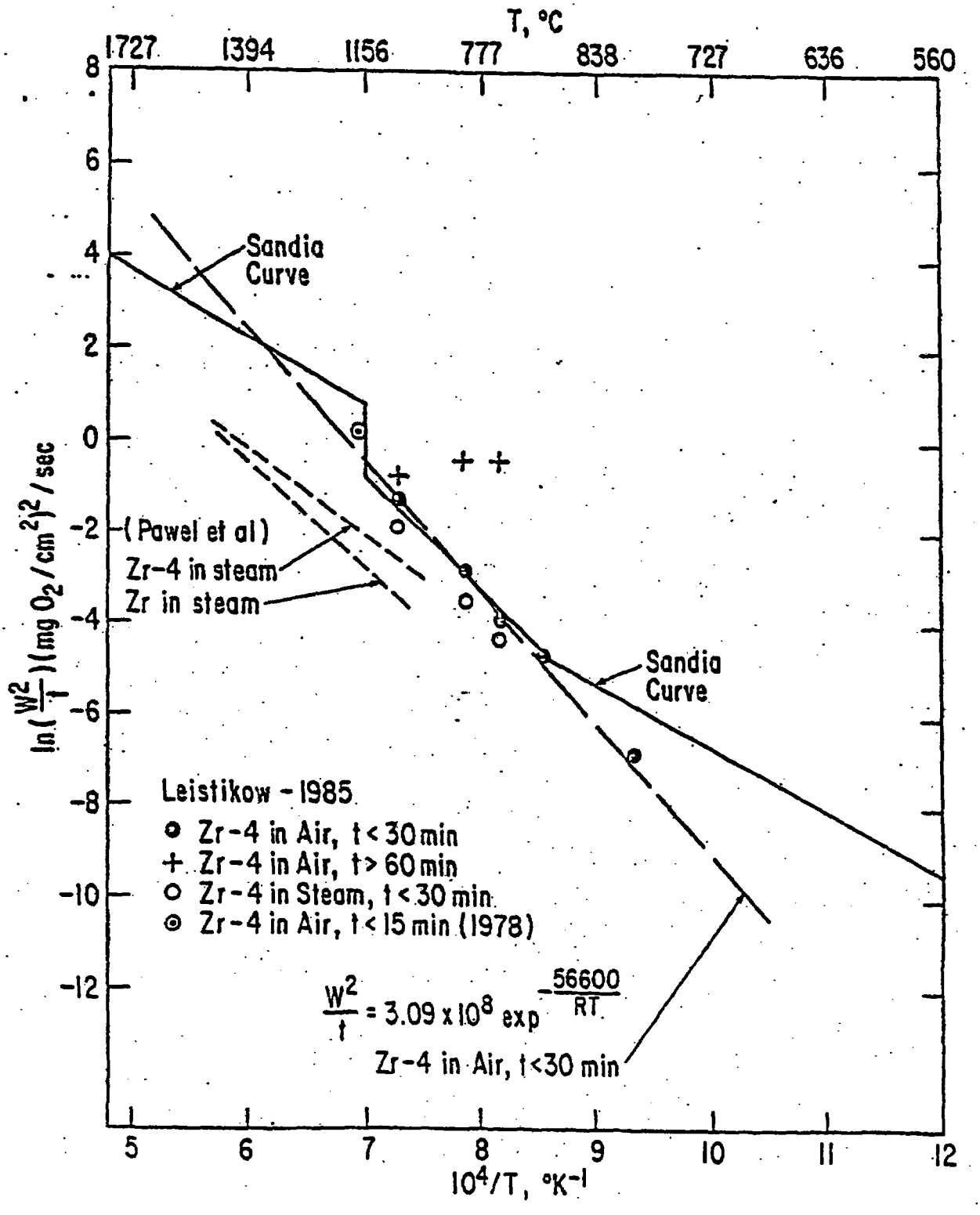

Figure B.3 Comparison of recent Zircaloy oxidation data with suggested correlations.

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APPENDIX C

EXAMPLE IMPACT FOR SFUEL AND CRAC2 CALCULATIONS

C.1 INTRODUCTION

As discussed in the main report the uncertainty, in the risk estimates for spent fuel pool accidents is dominated by the uncertainty in the frequency estimate for loss of pool integrity events. However, the phenomenological progression and the resulting source terms for such accidents also have large uncertainties. In order to provide documentation of the methods used to analyze the phenomena, typical input files for SFUEL[1,2] and CRAC2[3] are provided in this appendix.

C.2 SFUEL INPUT

The SFUEL code[1,2] was developed at Sandia National Laboratories (SNL) to address the problem of spent fuel pool heating after a loss-of-coolant event. The code has been used extensively in the previous investigations to address the conditions (geometry and power level) which would result in self-sustaining oxidation of the fuel rod cladding. The purpose of the present investigation is to provide an independent assessment of the SFUEL code and its ability to predict cladding oxidation and failure propagation. A review of the oxidation data is given in Appendix B and a summary of the propagation results is given in Section 3 of the main report. In order to ensure repeatability of the results, an input deck is provided in Figure C.1. The input corresponds to a PWR high density storage rack.

C.3 CRAC2 INPUT

CRAC2[3] has been used extensively to calculate off-site consequences for hypothetical core melt accidents. While its input options provide a wide range of user features which are subject to individual interpretation, in practice the input has become standardized. Unfortunately, these standardized features become less useful for spent fuel pool releases which tend to be dominated by long lived isotopes (principally cesium). In order to facilitate further sensitivity studies or possible plant specific analyses, the input file for a typical run (Case 2A) is provided in Figure C.2. Note that the fission product inventory has been changed from the "normal" assumption of an equilibrium core after shutdown to the release estimates given in Section 4 of the main report.

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```
 10 in,hd,lyr,vent

ASINK=   0.000E+00
CSINK =  0.000E+00
DAMP=    0.000E+00
DELT=    5.000E+01
DLFACT=  5.000E+00
DMWTR=   0.000E+00
         0.000E+00
         0.000E+00
FL=      3.550E+02
FSTR=    1.000E+00
IBLOCK=    3
ICHEM=     1
IPLOT=     0
N=        21
NCEND=     1
NPRINT=   36
NPRNEW=   36
NROD=    289
NSECT=     3
POWD=    4.614E-01
PRMAX=   1.150E+06
RCI=     4.180E-01
RCO=     4.750E-01
RP=      4.010E-01
ROWS=    0.000E+00
SMB=     2.500E-02
TIMAX=   6.000E+04
TIMWOF=  1.000E+09
TIMWON=  0.000E+00
TRDELT=  1.400E+03
TRMAX=   6.000E+03
TRPNT=   2.400E+03
UL=      0.000E+00
VENT=    0.000E+00
VROOM=   4.250E+09
WS=      2.160E+01
WW=      2.160E+01
XB=      6.350E-01
XL=      6.350E-01
XS=      0.000E+00
XTB=     4.060E+01
XW=      3.510E-01
XWL=     4.060E+01
XWW=     3.890E+00

  J  NASS        TIMEO

  1    4    0.000E+00
  2    4    0.000E+00
  3    4    0.000E+00

CPCON=   1.047E+00
CPL=     4.500E-01
CPNI=    1.130E+00
CPOX=    1.130E+00
CPS=     3.640E-01
CPW=     4.600E-01
EPC=     7.000E-01
```

Figure C.1 Input data for the SFUEL1W code corresponding to a PWR high density rack.

```
EPL=     3.000E-01
EPS=     7.000E-01
EPT=     7.000E-01
EPW=     3.000E-01
FMULT= -1.000E+00
KMAX=      15
NDECAY=     3
RHOC=    6.500E+00
RHOCON= 2.340E+00
RHOF=    1.040E+01
RHOL=    7.820E+00
RHOS=    6.500E+00
RHOW=    7.820E+00
SMKCON= 1.200E-02
TO=      2.830E+02
XKBOT=  1.650E+02
XKTOP=  0.000E+00
```

I	TCOOL	FDECAY
1	1.000E-01	4.000E+00
2	1.000E+00	5.900E+00
3	1.000E+01	1.100E+01

Figure C.1 (Cont'd)

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```
TWENTY FIVE REGIONS
SPATIAL                    25     NO
.5        1.0        1.5        2.         2.5        3.0        3.5        4.
4.5       5.         6.         7.         8.5        10.        12.5       15.
17.5      20.        25.        30.        35.        40.        45.        50.
55.
SITE                       1
INDPT WITH NYC MET                 5 0 0 0 1
   29   04
ECONOMIC                   1.                NO
  499.0  3349.0        0.2      31527.0    4344.0    135.0       885.0
N.Y.          5    9 0.315      0.579      188.0     842.0
POPULATION                 .16
  2
  UNIFORM POPULATION AND WIND ROSE
  6.25E-02  4.91E+00  1.47E+01  2.46E+01  3.44E+01  4.42E+01  5.40E+01  6.38E+0
  7.37E+01  8.35E+01  9.33E+01  2.16E+02  2.55E+02  4.57E+02  5.45E+02  1.10E+0
  1.35E+03  1.60E+03  1.84E+03  4.42E+03  5.40E+03  6.38E+03  7.37E+03  8.35E+0
  9.33E+03  1.03E+04
  6.25E-02  4.91E+00  1.47E+01  2.46E+01  3.44E+01  4.42E+01  5.40E+01  6.38E+0
  7.37E+01  8.35E+01  9.33E+01  2.16E+02  2.55E+02  4.57E+02  5.45E+02  1.10E+0
  1.35E+03  1.60E+03  1.84E+03  4.42E+03  5.40E+03  6.38E+03  7.37E+03  8.35E+0
  9.33E+03  1.03E+04
  6.25E-02  4.91E+00  1.47E+01  2.46E+01  3.44E+01  4.42E+01  5.40E+01  6.38E+0
  7.37E+01  8.35E+01  9.33E+01  2.16E+02  2.55E+02  4.57E+02  5.45E+02  1.10E+0
  1.35E+03  1.60E+03  1.84E+03  4.42E+03  5.40E+03  6.38E+03  7.37E+03  8.35E+0
  9.33E+03  1.03E+04
  6.25E-02  4.91E+00  1.47E+01  2.46E+01  3.44E+01  4.42E+01  5.40E+01  6.38E+0
  7.37E+01  8.35E+01  9.33E+01  2.16E+02  2.55E+02  4.57E+02  5.45E+02  1.10E+0
  1.35E+03  1.60E+03  1.84E+03  4.42E+03  5.40E+03  6.38E+03  7.37E+03  8.35E+0
  9.33E+03  1.03E+04
  6.25E-02  4.91E+00  1.47E+01  2.46E+01  3.44E+01  4.42E+01  5.40E+01  6.38E+0
  7.37E+01  8.35E+01  9.33E+01  2.16E+02  2.55E+02  4.57E+02  5.45E+02  1.10E+0
  1.35E+03  1.60E+03  1.84E+03  4.42E+03  5.40E+03  6.38E+03  7.37E+03  8.35E+0
  9.33E+03  1.03E+04
  6.25E-02  4.91E+00  1.47E+01  2.46E+01  3.44E+01  4.42E+01  5.40E+01  6.38E+0
  7.37E+01  8.35E+01  9.33E+01  2.16E+02  2.55E+02  4.57E+02  5.45E+02  1.10E+0
  1.35E+03  1.60E+03  1.84E+03  4.42E+03  5.40E+03  6.38E+03  7.37E+03  8.35E+0
  9.33E+03  1.03E+04
  6.25E-02  4.91E+00  1.47E+01  2.46E+01  3.44E+01  4.42E+01  5.40E+01  6.38E+0
  7.37E+01  8.35E+01  9.33E+01  2.16E+02  2.55E+02  4.57E+02  5.45E+02  1.10E+0
  1.35E+03  1.60E+03  1.84E+03  4.42E+03  5.40E+03  6.38E+03  7.37E+03  8.35E+0
  9.33E+03  1.03E+04
  6.25E-02  4.91E+00  1.47E+01  2.46E+01  3.44E+01  4.42E+01  5.40E+01  6.38E+0
  7.37E+01  8.35E+01  9.33E+01  2.16E+02  2.55E+02  4.57E+02  5.45E+02  1.10E+0
  1.35E+03  1.60E+03  1.84E+03  4.42E+03  5.40E+03  6.38E+03  7.37E+03  8.35E+0
  9.33E+03  1.03E+04
  6.25E-02  4.91E+00  1.47E+01  2.46E+01  3.44E+01  4.42E+01  5.40E+01  6.38E+0
  7.37E+01  8.35E+01  9.33E+01  2.16E+02  2.55E+02  4.57E+02  5.45E+02  1.10E+0
  1.35E+03  1.60E+03  1.84E+03  4.42E+03  5.40E+03  6.38E+03  7.37E+03  8.35E+0
  9.33E+03  1.03E+04
  6.25E-02  4.91E+00  1.47E+01  2.46E+01  3.44E+01  4.42E+01  5.40E+01  6.38E+0
  7.37E+01  8.35E+01  9.33E+01  2.16E+02  2.55E+02  4.57E+02  5.45E+02  1.10E+0
  1.35E+03  1.60E+03  1.84E+03  4.42E+03  5.40E+03  6.38E+03  7.37E+03  8.35E+0
  9.33E+03  1.03E+04
  6.25E-02  4.91E+00  1.47E+01  2.46E+01  3.44E+01  4.42E+01  5.40E+01  6.38E+0
```

Figure C.2 Input data for the CRAC2 code for spent fuel release corresponding to 1/3 core after 90 days decay (Case 2A in Table 4.7.

```
 7.37E+01  8.35E+01  9.33E+01  2.16E+02  2.55E+02  4.57E+02  5.45E+02  1.10E+0
 1.35E+03  1.60E+03  1.84E+03  4.42E+03  5.40E+03  6.38E+03  7.37E+03  8.35E+0
 9.33E+03  1.03E+04
 6.25E-02  4.91E+00  1.47E+01  2.46E+01  3.44E+01  4.42E+01  5.40E+01  6.38E+0
 7.37E+01  8.35E+01  9.33E+01  2.16E+02  2.55E+02  4.57E+02  5.45E+02  1.10E+0
 1.35E+03  1.60E+03  1.84E+03  4.42E+03  5.40E+03  6.38E+03  7.37E+03  8.35E+0
 9.33E+03  1.03E+04
 6.25E-02  4.91E+00  1.47E+01  2.46E+01  3.44E+01  4.42E+01  5.40E+01  6.38E+0
 7.37E+01  8.35E+01  9.33E+01  2.16E+02  2.55E+02  4.57E+02  5.45E+02  1.10E+0
 1.35E+03  1.60E+03  1.84E+03  4.42E+03  5.40E+03  6.38E+03  7.37E+03  8.35E+0
 9.33E+03  1.03E+04
 6.25E-02  4.91E+00  1.47E+01  2.46E+01  3.44E+01  4.42E+01  5.40E+01  6.38E+0
 7.37E+01  8.35E+01  9.33E+01  2.16E+02  2.55E+02  4.57E+02  5.45E+02  1.10E+0
 1.35E+03  1.60E+03  1.84E+03  4.42E+03  5.40E+03  6.38E+03  7.37E+03  8.35E+0
 9.33E+03  1.03E+04
 6.25E-02  4.91E+00  1.47E+01  2.46E+01  3.44E+01  4.42E+01  5.40E+01  6.38E+0
 7.37E+01  8.35E+01  9.33E+01  2.16E+02  2.55E+02  4.57E+02  5.45E+02  1.10E+0
 1.35E+03  1.60E+03  1.84E+03  4.42E+03  5.40E+03  6.38E+03  7.37E+03  8.35E+0
 9.33E+03  1.03E+04
TOPOG                  0
  110
ISOTOPE                54    ND
CO-58     7            1.26 E+03 7.130E+01 1.000E-02 1.000E-00
CO-60     7            8.68 E+03 1.921E+03 1.000E-02 1.000E-00
KR-85     1            2.36 E+05 3.919E+03 0.        0.
KR-85M    1            0.00      1.867E-01 0.        0.
KR-87     1            0.00      5.278E-02 0.        0.
KR-88     1            0.00      1.167E-01 0.        0.
RB-86     4            1.05 E+03 1.865E+01 1.000E-02 1.000E-00
SR-89     6            7.75 E+03 5.200E+01 1.000E-02 1.000E-00
SR-90     6            3.82 E+03 1.026E+04 1.000E-02 1.000E-00
SR-91     6            0.00      3.950E-01 1.000E-02 1.000E-00
Y-90      8 SR-90      3.84 E+03 2.670E+00 1.000E-02 1.000E-00
Y-91      8 SR-91      1.30 E+04 5.880E+01 1.000E-02 1.000E-00
ZR-95     8            8.37 E+05 6.550E+01 1.000E-02 1.000E-00
ZR-97     8            0.00      7.000E-01 1.000E-02 1.000E-00
NB-95     8 ZR-95      1.41 E+06 3.510E+01 1.000E-02 1.000E-00
MO-99     7            0.00      2.751E+00 1.000E-02 1.000E-00
TC-99M    7 MO-99      0.00      2.508E-01 1.000E-02 1.000E-00
RU-103    7            1.04 E+02 3.959E+01 1.000E-02 1.000E-00
RU-105    7            0.00      1.850E-01 1.000E-02 1.000E-00
RU-106    7            1.99 E+02 3.690E+02 1.000E-02 1.000E-00
RH-105    7 RU-105     0.00      1.479E+00 1.000E-02 1.000E-00
SB-127    5            1.39 E-01 3.800E+00 1.000E-02 1.000E-00
SB-129    5            0.00      1.808E-01 1.000E-02 1.000E-00
TE-127    5 SB-127     2.80 E+03 3.896E-01 1.000E-02 1.000E-00
TE-127M   5            2.86 E+03 1.090E+02 1.000E-02 1.000E-00
TE-129    5 SB-129     1.56 E+03 4.861E-02 1.000E-02 1.000E-00
TE-129M   5            2.40 E+03 3.340E+01 1.000E-02 1.000E-00
TE-131M   5            0.00      1.250E+00 1.000E-02 1.000E-00
TE-132    5            1.73 E-03 3.250E+00 1.000E-02 1.000E-00
I-131     3 TE-131M    6.35 E+03 8.040E+00 1.000E-02 1.000E-00
I-132     3 TE-132     1.78 E-03 9.521E-02 1.000E-02 1.000E-00
I-133     3            0.00      8.667E-01 1.000E-02 1.000E-00
I-134     3            0.00      3.653E-02 1.000E-02 1.000E-00
I-135     3            0.00      2.744E-01 1.000E-02 1.000E-00
XE-133    1 I-133      2.30 E+02 5.290E+00 0.        0.
XE-135    1 I-135      0.00      3.821E-01 0.        0.
CS-134    4            3.34 E+06 7.524E+02 1.000E-02 1.000E-00
CS-136    4            8.13 E+03 1.300E+01 1.000E-02 1.000E-00
```

Figure C.2 (Cont'd)

```
CS-137    4            2.81 E+06 1.099E+04 1.000E-02 1.000E-00
BA-140    6            3.80 E+03 1.279E+01 1.000E-02 1.000E-00
LA-140    8 BA-140     4.38 E+02 1.676E+00 1.000E-02 1.000E-00
CE-141    8            3.61 E+00 8.253E+01 1.000E-02 1.000E-00
CE-143    8            0.00      1.375E+00 1.000E-02 1.000E-00
CE-144    8            1.65 E+01 2.844E+02 1.000E-02 1.000E-00
PR-143    8 CE-143     2.41 E-01 1.358E+01 1.000E-02 1.000E-00
ND-147    8            3.36 E-02 1.099E+01 1.000E-02 1.000E-00
NP-239    8            0.00      2.350E+00 1.000E-02 1.000E-00
PU-238    8 CM-242     6.87 E-02 3.251E+04 1.000E-02 1.000E-00
PU-239    8 NP-239     9.28 E-03 8.912E+06 1.000E-02 1.000E-00
PU-240    8 CM-244     1.55 E-02 2.469E+06 1.000E-02 1.000E-00
PU-241    8            3.70 E+00 5.333E+03 1.000E-02 1.000E-00
AM-241    8 PU-241     7.00 E-03 1.581E+05 1.000E-02 1.000E-00
CM-242    8            1.01 E+00 1.630E+02 1.000E-02 1.000E-00
CM-244    8            5.84 E-02 6.611E+03 1.000E-02 1.000E-00
LEAKAGE                  1    NO
EARLY     1.0       0.0       5.0       0.0          1.4E05 80.0
 1.0      0.0       1.0       1.0       1.0      1.0       1.0      1.0
DISPERSION
100.0     25.0          0   34    0
EVACUATE                 1    NO        NO
      1.0                                                        2.        .2
      .75        1.        .5       .75       .33       .5       .08       .3
   2.66E-4   2.66E-4   1.33E-4   2.66E-4
    8045.       90.       165.        3.    1
ACUTE                    7
T MARROW  320.      400.      510.      615.      .03       .5        1.
LLI WALL  2000.     5000.     5000.     5000.     1.        1.        1.
LUNG      5000.     14800.    22400.    24000.    .24       .73       1.
W BODY    55.       150.      280.      870.      .30       .8        0.
LUNG      3000.     3000.1    6000.     6000.     .05       1.0       0.
LLI WALL  1000.     1000.1    2500.     2500.     .05       1.0       0.
THYROID        1.E10     1.E10     1.E10     1.E10     1.0       1.0       0.
LATENT                   8
   10CENT EST  80.        5.        300.      2.5
T MARROW  LEUKEMIA   2.836E-05 2.720E-05 1.872E-05 1.382E-05 9.720E-06 6.770E-0
 4.040E-06 1.700E-06 4.900E-07 0.0       1.0
LUNG      LUNG       2.749E-05 2.749E-05 2.749E-05 1.587E-05 8.130E-06 3.990E-0
 1.500E-06 2.200E-07 0.0       0.0       0.5
OTHER     BREAST     3.172E-05 3.172E-05 3.172E-05 1.831E-05 9.380E-06 4.600E-0
 1.730E-06 2.500E-07 0.0       0.0       1.000E+09
SKELETON  BONE       1.107E-05 1.064E-05 6.990E-06 3.020E-06 1.670E-06 9.100E-0
 4.200E-07 1.200E-07 1.000E-08 0.0       1.0
LLI WALL  GI TRK     1.688E-05 1.688E-05 1.688E-05 9.740E-06 4.990E-06 2.450E-0
 9.200E-07 1.300E-07 0.0       0.0       1.0
OTHER     OTHER      3.220E-05 3.050E-05 2.539E-05 1.466E-05 7.520E-06 3.690E-0
 1.390E-06 2.000E-07 0.0       0.0       1.0
W BODY    W BODY     1.579E-04 1.533E-04 1.274E-04 7.542E-05 4.141E-05 2.241E-0
 1.000E-05 2.620E-06 5.000E-07 0.0       1.0
THYROID   THYROID     3.34E-04
                                          1.00E 09
CHRONIC EXPOSURE         6
   10    1     1.000      365.     25550.                    3.0       15.0
SR-90          0.0525    0.0718
RU-106         0.0397    0.0533
CS-137         0.0525    0.105
PU-238         0.0529    0.107
PU-239         0.0530    0.108
PU-240         0.0530    0.108
```

Figure C.2 (Cont'd)

PU-241		0.0520	0.101				
AM-241		0.0530	0.108				
CM-242		0.0292	0.0327				
CM-244		0.0522	0.102				
3	11	1.0	365.	365.	14.0	2.0	3.3
CS-134		8.44	4.22				
LUNG		6.47E+4	7.31E+4				
T MARROW		6.50E+4	7.34E+4				
SKELETON		6.41E+4	7.24E+4				
T E C L		6.41E+4	7.24E+4				
ST WALL		7.40E+4	8.34E+4				
SI+CONT		8.05E+4	9.09E+4				
ULI WALL		7.95E+4	8.96E+4				
LLI WALL		8.28E+4	9.33E+4				
THYROID		6.49E+4	7.33E+4				
OTHER		6.27E+4	7.08E+4				
W BODY		6.32E+4	7.14E+4				
TESTES		7.57E+4	8.55E+4				
OVARIES		6.68E+4	7.55E+4				
CS-136		2.84	1.42				
LUNG		8.82E+3	8.82E+3				
T MARROW		9.29E+3	9.29E+3				
SKELETON		9.10E+3	9.10E+3				
T E C L		9.10E+3	9.10E+3				
ST WALL		1.15E+4	1.15E+4				
SI+CONT		1.19E+4	1.19E+4				
ULI WALL		1.20E+4	1.20E+4				
LLI WALL		1.35E+4	1.35E+4				
THYROID		9.23E+3	9.23E+3				
OTHER		8.88E+3	8.88E+3				
W BODY		8.96E+3	8.96E+3				
TESTES		1.03E+4	1.03E+4				
OVARIES		9.48E+3	9.48E+3				
CS-137		8.44	4.22				
LUNG		4.71E+4	5.59E+4				
T MARROW		4.73E+4	5.61E+4				
SKELETON		4.68E+4	5.56E+4				
T E C L		4.68E+4	5.56E+4				
ST WALL		5.18E+4	6.13E+4				
SI+CONT		5.39E+4	6.39E+4				
ULI WALL		5.40E+4	6.39E+4				
LLI WALL		5.64E+4	6.64E+4				
THYROID		4.68E+4	5.55E+4				
OTHER		4.60E+4	5.45E+4				
W BODY		4.62E+4	5.49E+4				
TESTES		5.18E+4	6.15E+4				
OVARIES		4.81E+4	5.70E+4				
2	2	1.0	365.	365.	14.0	2.0	3.3
SR-89		.397	0.402				
LUNG		2.91E+3	5.81E+2				
T MARROW		2.63E+4	5.26E+3				
SKELETON		5.95E+4	1.19E+4				
T E C L		6.00E+4	1.20E+4				
ST WALL		1.56E+4	3.12E+3				
SI+CONT		2.73E+4	5.45E+3				
ULI WALL		1.46E+5	2.91E+4				
LLL WALL		4.27E+5	8.53E+4				
THYROID		2.91E+3	5.81E+2				
OTHER		2.91E+3	5.81E+2				
W BODY		9.55E+3	1.91E+3				

Figure C.2 (Cont'd)

000375

TESTES		2.91E+3	5.81E+2				
OVARIES		2.91E+3	5.81E+2				
SR-90		.505	.588				
LUNG		1.59E+4	3.18E+3	5.50E+2	1.80E+1		
T MARROW		1.04E+6	2.08E+5	5.25E+4	1.29E+4	1.00E+4	3.10E+3
SKELETON		3.08E+6	6.15E+5	2.57E+5	9.81E+4	1.09E+5	4.30E+4
T E C L		2.64E+6	5.27E+5	1.93E+5	6.77E+4	7.20E+4	2.76E+4
ST WALL		2.03E+4	4.05E+3	5.50E+1	1.80E+1		
SI+CONT		2.64E+4	5.28E+3	5.50E+1	1.80E+1		
ULI WALL		1.06E+5	2.11E+4	5.00E+1	2.00E+1		
LLI WALL		4.06E+5	8.12E+4	5.00E+1			
THYROID		1.59E+4	3.18E+3	5.50E+1	1.80E+1		
OTHER		1.59E+4	3.18E+3	5.50E+1	1.30E+1		
W BODY		2.76E+5	5.52E+4	2.03E+4	7.44E+3	8.08E+3	3.13E+3
TESTES		1.59E+4	3.18E+3	5.50E+1	1.80E+1		
OVARIES		1.59E+4	3.18E+3	5.50E+1	1.80E+1		
2	9	1.0	0.0	385.	-14.0	0.0	10.0
I-133		1.00E-8	0.00486				
LUNG		8.53E+2	1.58E+2				
T MARROW		7.99E+2	1.48E+2				
SKELETON		7.88E+2	1.46E+2				
T E C L		7.88E+2	1.46E+2				
ST WALL		1.11E+4	2.06E+3				
SI+CONT		2.30E+3	4.25E+2				
ULI WALL		6.16E+3	1.14E+3				
LLI WALL		9.83E+3	1.82E+3				
THYROID		1.73E+6	3.21E+5				
OTHER		9.07E+2	1.68E+2				
W BODY		1.46E+3	2.70E+2				
TESTES		7.13E+2	1.32E+2				
OVARIES		9.99E+2	1.85E+2				
I-131		1.00E-8	0.692				
LUNG		1.92E+3	3.56E+2				
T MARROW		1.55E+3	2.87E+2				
SKELETON		1.67E+3	3.10E+2				
T E C L		1.67E+3	3.10E+2				
ST WALL		6.16E+3	1.14E+3				
SI+CONT		1.79E+3	3.32E+2				
ULI WALL		4.49E+3	8.32E+2				
LLI WALL		1.03E+4	1.91E+3				
THYROID		9.07E+6	1.68E+6				
OTHER		2.20E+3	4.07E+2				
W BODY		4.75E+3	8.79E+2				
TESTES		7.51E+3	1.39E+2				
OVARIES		1.15E+3	2.21E+2				
4	2	1.000	3650.	3650.	2400.	5.0	5.0
CS-134		.184	.0547				
LUNG		7.31E+4					
T MARROW		7.34E+4					
SKELETON		7.24E+4					
T E C L		7.24E+4					
ST WALL		8.34E+4					
SI+CONT		9.09E+4					
ULI WALL		8.96E+4					
LLI WALL		9.33E+4					
THYROID		7.33E+4					
OTHER		7.08E+4					
W BODY		7.14E+4					
TESTES		8.55E+4					
OVARIES		7.55E+4					

Figure C.2 (Cont'd)

CS-137	.250	.0835				
LUNG	5.59E+4					
T MARROW	5.61E+4					
SKELETON	5.56E+4					
T E C L	5.56E+4					
ST WALL	6.13E+4					
SI+CONT	6.39E+4					
ULI WALL	6.39E+4					
LLI WALL	6.64E+4					
THYROID	5.55E+4					
OTHER	5.45E+4					
W BODY	5.49E+4					
TESTES	6.15E+4					
OVARIES	5.70E+4					
SR-89	.0136	.0068				
LUNG	5.81E+2					
T MARROW	5.26E+3					
SKELETON	1.19E+4					
T E C L	1.20E+4					
ST WALL	3.12E+3					
SI+CONT	5.45E+3					
ULI WALL	2.91E+4					
LLL WALL	8.53E+4					
THYROID	5.81E+2					
OTHER	5.81E+2					
W BODY	1.91E+3					
TESTES	5.81E+2					
OVARIES	5.81E+2					
SR-90	1.340	0.669				
LUNG	3.18E+3	5.50E+2	1.80E+1			
T MARROW	2.08E+5	5.25E+4	1.29E+4	1.00E+4	3.10E+3	3.103+3
SKELETON	6.15E+5	2.57E+5	9.81E+4	1.09E+5	4.30E+4	4.30E+4
T E C L	5.27E+5	1.93E+5	6.77E+4	7.20E+4	2.76E+4	2.76E+4
ST WALL	4.05E+3	5.50E+1	1.80E+1			
SI+CONT	5.28E+3	5.50E+1	1.80E+1			
ULI WALL	2.11E+4	5.00E+1	2.00E+1			
LLL WALL	8.12E+4	5.00E+1				
THYROID	3.18E+3	5.50E+1	1.80E+1			
OTHER	3.18E+3	5.50E+1	1.30E+1			
W BODY	5.52E+4	2.03E+4	7.44E+3	8.08E+3	3.13E+3	3.13E+3
TESTES	3.18E+3	5.50E+1	1.80E+1			
OVARIES	3.18E+3	5.50E+1	1.80E+1			
10 11	0.333	365.	10950.	3285.	5.0	25.0
CO-58						
CO-60						
NB-95						
ZR-95						
RU-103						
RU-106						
I-131						
CS-134						
CS-136						
CS-137						
SCALE		36	NO			
1.	E02.	E03.	E05.	E07.	E01.	E12.	E13.	E
5.	E17.	E11.	E22.	E23.	E25.	E27.	E21.	E
2.	E33.	E35.	E37.	E31.	E42.	E43.	E45.	E
7.	E41.	E52.	E53.	E55.	E57.	E51.	E62.	E
3.	E65.	E67.	E61.	E7				
RESULT		19	NO					

Figure C.2 (Cont'd)

000357

```
ACUTE FATALITIES
ACUTE INJURIES
FATAL RADIUS(MI)
INJUR RADIUS(MI)
ACU BMR DS-INT 2                    1.0E-02
ACU BMR DS-INT10                    1.0E-02
ACU BMR DS-INT14                    1.0E-02
ACU BMR DS-INT18                    1.0E-02
ACU BMR DS-INT20                    1.0E-02
ACU BMR DS-INT24                    1.0E-02
ACU THY DS-INT 2                    1.0E-02
ACU THY DS-INT10                    1.0E-02
ACU THY DS-INT14                    1.0E-02
ACU THY DS-INT18                    1.0E-02
ACU THY DS-INT20                    1.0E-02
ACU THY DS-INT24                    1.0E-02
TOT WBODY MANREM                    1.0E+02
INTERD AREA
DECON AREA
OPTIONS                        NO
                  -1
END
 RUN NO.5, CASE2A, 90-DAY DECAY, 100% RELEASE
EVACUATE                 1
       1.0                                                                  1.0
       .67        1.0        .5        .87        .32        .5        .05   .32
   2.66E-4   2.66E-4   1.33E-4   2.66E-4
     8045.       90.        95.        3.    1
END
```

Figure C.2 (Cont'd)

000358

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

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1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)

NUREG/CR-4982
BNL-NUREG-52093

2. TITLE AND SUBTITLE

Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
June	1987

5. AUTHOR(S)

V.L. Sailor, K.R. Perkins, J.R. Weeks and H.R. Connell

6. DATE REPORT ISSUED

MONTH	YEAR
July	1987

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Brookhaven National Laboratory
Upton, NY 11973

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER

FIN A-3786

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Division of Reactor and Plant Systems
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

11. TYPE OF REPORT

12. PERIOD COVERED (Inclusive dates)

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This investigation provides an assessment of the likelihood and consequences of a severe accident in a spent fuel storage pool — the complete draining of the pool. Potential mechanisms and conditions for failure of the spent fuel, and the subsequent release of the fission products, are identified. Two older PWR and BWR spent fuel storage pool designs are considered based on a preliminary screening study which tried to identify vulnerabilities. Internal and external events and accidents are assessed. Conditions which could lead to failure of the spent fuel Zircaloy cladding as a result of cladding rupture or as a result of a self-sustaining oxidation reaction are presented. Propagation of a cladding fire to older stored fuel assemblies is evaluated. Spent fuel pool fission product inventory is estimated and the releases and consequences for the various cladding failure scenarios are provided. Possible preventive or mitigative measures are qualitatively evaluated. The uncertainties in the risk estimate are large, and areas where additional evaluations are needed to reduce uncertainty are identified.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS

Spent fuel pools, generic safety issue, probabilistic risk assessment, PWRs, BWRs, Zircaloy oxidation, seismic hazard, fragility fission products, consequences, failure propagation, oxidation rate, SFUEL, CRAC2, ORIGEN2, decay heat

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT

Unlimited

16. SECURITY CLASSIFICATION

(This page)
Unclassified

(This report)
Unclassified

17. NUMBER OF PAGES

18. PRICE

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Environment

Indian Point chief: Plant safe from possible attack

By ROGER WITHERSPOON
THE JOURNAL NEWS
(Original publication: Oct. 20, 2001)

If a wide-bodied jet were to fly into one of Indian Point's two nuclear power plants, it is unlikely the crash would cause enough damage to trigger a nuclear catastrophe, the chief executive of the company that owns the plants said yesterday.

Mike Kansler, head of Entergy Nuclear Northeast, said that the plants' massive, concrete containment buildings and internal barriers should prevent a major incident. But he acknowledged that there is no evidence to substantiate that belief.

"People are worrying about a plane crashing through," Kansler said during a meeting with The Journal News Editorial Board. "While it is true they were not specifically designed, and we have not done the calculations to say that these plants could specifically withstand a 767 or 747 hitting it, our belief is because of the robustness of the design, the way they were built, the way the reactors were placed in the building, we believe the plant could survive such an incident."

He added that there are several barriers within the containment buildings protecting the nuclear reactor and its 30 tons of fuel. A plane, he said, "may be able to penetrate the containment building, but not the rest."

Kansler said that in the wake of the Sept. 11 attacks on the World Trade Center, there was a public perception that the two nuclear plants at Indian Point, and their huge pools of spent nuclear fuel, were potential targets for a similar suicide attack.

Kansler said he did not believe the plants at Indian Point were a tempting terrorist target. In the worst case scenario, a serious accident at the plants in Buchanan could affect more than 20 million people within a 50-mile radius in New York, New Jersey, Connecticut, and Pennsylvania.

"They are not a preferred target," Kansler said. "A plane did fly down the Hudson River on its way to the World Trade Center. But the World

Trade Center is a much neater target for a plane to go after. It has much more of a public impact than a nuclear plant."

The U.S. Nuclear Regulatory Commission recently acknowledged that such plants were never designed to withstand an attack by a fuel-laden, wide-bodied jet.

"We have not done the analysis, so we are not going to guarantee that a plane of that nature couldn't breach the containment," NRC spokesman Neil Sheehan said yesterday. "We are quick to add that these are very hardened structures, they have a lot of systems in place to shut down the reactor in the event of an attack like that."

Kansler and other Entergy officials would not specify what security measures were in place at Indian Point, citing safety concerns. But they said the measures did take into account the possibility of an aerial attack. Kansler said he did not believe anti-aircraft missiles were the best way to protect plants, and questioned how it would be determined that a commercial airplane was headed toward the plants.

He and other company officials also said that the NRC and federal government also were responsible for helping to protect nuclear plants during times of war, whether officially declared or not.

Kansler said that if a plane crashed into one of the containment domes and did not cause a nuclear meltdown, "it is not going to have the devastation and disruptive impact that you get hitting the World Trade Center or hitting the Pentagon. And we believe the plant is viable and it could withstand such an attack."

The Coast Guard has been providing protection along the Hudson River since Sept. 11, and had announced its intention to end its patrols Monday. That decision prompted state and federal officials from Westchester to demand that the 24-hour surveillance continue. Gov. George Pataki, who visited Indian Point yesterday, announced that the Coast Guard would maintain its 24-hour patrols.

Kansler said he was not concerned about the presence or departure of the Coast Guard vessels.

"We can deal with a security threat, no matter what kind it is," he said.

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Nuclear plant scare adds to U.S. security jitters
Thursday, October 18, 2001 04:44 PM ET

By Vibeke Laroi

SAN FRANCISCO, (Reuters) - A security scare at the Three Mile Island power plant sent jitters through the U.S. nuclear power industry Thursday, driving

home the urgent need to rewrite outdated rules to met new threats following the Sept. 11 attacks.

Those attacks on the World Trade Center and Pentagon have opened up a previously unthinkable, possibility: using a big airliner to smash into a nuclear power station.

"No one considered the possibility of suicide hijackers steering a large aircraft into a nuclear plant," said Victor Dricks, a spokesman with the Nuclear Regulatory Commission (NRC), which oversees the nuclear industry.

"That was not considered ... a 'credible threat'. That's an act of war. Some

senarios that we previously did not consider credible may now need to be considered."

Under current NRC regulations, a plant has to be able to defend itself against a small group of terrorists attacking from the ground with inside help.

As part of an ongoing comprehensive review of its regulations, the Washington-based NRC shut down its Web site to rethink its policy of posting

thousands of documents.

For each of the country's 103 nuclear power reactors, the NRC had put on its

Web site engineering drawings, photographs of the plants and detailed technical data describing the function of dozens of safety-related systems.

There were also aerial photographs of portions of sites, "which clearly

could
be of use to potential terrorists planning an aerial assault," Dricks said.

The NRC has decided to permanently cut from its Web site the latitudes and longitudes pinpointing the positions of all U.S. nuclear plants, he added.

THREATS

On Thursday, U.S. officials dismissed an unspecified threat to the Three Mile
Island nuclear power station as no longer credible, less than a day after putting the Pennsylvania plant on high alert and shutting two nearby airports.

The nation's 103 commercial reactors were already on heightened alert following the Sept. 11 attacks.

A few intrusions on nuclear plants have been reported, like in 1993 when a car, driven by an person with a history of mental illness, crashed through the outer gates of the Three Mile Island compound.

But a U.S. nuclear reactor has never been attacked.

The NRC says reactors, with their robust steel-enforced concrete containment

buildings and multiple safety systems, are, "among the most hardened structures in the country designed to withstand hurrianes, tornadoes and earthquakes.

"However, we could not rule out the possibility that a large aircraft like a

(Boeing) 767, fully loaded with fuel, deliberately crashed into a containment
building might not cause some structural damage that could release some radiation," Dricks said.

He added detailed engineering analyses of a large airliner crash on a nuclear
plant have not yet been performed.

But Paul Leventhal, president of watchdog group Nuclear Control Institute (NCI) warned that a Boeing 767 flown at full speed could penetrate a reactor

containment vessel, causing a meltdown and release of radiation.

"The consequences are so unthinkable that a successful attack has to be prevented at all costs," he told Reuters.

The NCI is calling on the government to immediately station National Guard

soldiers and anti-aircraft weapons around each nuclear plant to boost their defense.

"We sent letters to the governors of 40 states, all of which have nuclear plants, urging them to establish close communication with the National Guard

within their borders in the event that they need assistance. So that's been left in their hands," Dricks said.

So far, New York and New Jersey have called up National Guard troops to protect the plants.

"We have taken steps to ensure the plants will be able to defend themselves against a wider variety of attacks ... and that includes attacks from the air
and water," Dricks said, without going into details.

The Federal Aviation Administration (FAA) has also issued an advisory to pilots urging them to stay away from nuclear plants, regardless of whether they are in their flight path.

Nation & World 9/17/01

A nuclear nightmare

BY DOUGLAS PASTERNAK

He called it Project Worst Nightmare. And in the twisted mind of Donald Beauregard, commander of the 77th Regiment Militia in St. Petersburg, Fla., it surely was. Beauregard's plan was simple–disable the electric power grid feeding the nearby Crystal River nuclear power plant with explosives stolen from a National Guard armory. That would shut down the plant, blacking out St. Petersburg. This was no idle fantasy. When the cops finally caught up with him, Beauregard and his "strike team" had a 20-mm cannon, a .50-caliber machine gun, and a few pipe bombs primed to blow.

Beauregard might have succeeded if an informant hadn't tipped the police. He was prosecuted and clapped off to prison last year. But the FBI took Beauregard's plan seriously enough to incorporate it into a test it ran last May against the Palo Verde nuclear generating station in Arizona.

And here lies the rub. In the past decade, nearly half the nation's 103 power plants have failed mock terrorist attacks against them. The plants that failed, in other words, would not have stopped the Donald Beauregards of the world.

In the parlance of counterterrorism, nuclear power plants are among the world's most "hardened" targets. Barbed wire, surveillance cameras, motion sensors, armed response teams–all are designed to make the plants impenetrable to even the most determined saboteur. But interviews with current and former Nuclear Regula- tory Commission inspectors, security experts, and plant guards paint a very different picture. Often, security measures at nuclear plants don't work as they should or don't work at all. A re- view of recent incidents by *U.S. News* reveals numerous breakdowns in plant security, from criminals being granted access to sensitive areas to inadequate security that places vital equipment within easy reach of an attacker who never even enters the plant's perimeter.

Security experts say a terrorist is far more likely to attack a so-called soft target– such as a government building–than a nuclear power plant. Indeed, argues Lynnette Hendricks of the Nuclear Energy Institute, the nuclear power trade group: "We believe the plants are overly defended at a level that is not at all commensurate with the risk." But in light of attacks against fortified targets such as U.S. embassies, threats against nuclear plants are now considered very real. And concerns about security are likely to mount as the Bush administration calls for greater use of nuclear power. Last year, for instance, Japanese police arrested a man with seven pipe bombs who was planning to blow up a uranium processing plant. Last September,

Ukrainian police arrested a group planning to sabotage the Chernobyl reactor. And in the United States, officials list at least 30 threats against nuclear plants since 1978. Most have been hoaxes, but in the mid-1980s, for instance, three of four power lines leading to the Palo Verde plant were sabotaged. And in 1989 four members of Earth First!, a radical environmental group, were charged with conspiring to disable three nuclear power plants in the Southwest.

Rating risks. Despite the threats and the documented security flaws, the nuclear industry has convinced the Nuclear Regulatory Commission–the federal agency that oversees nuclear power plants–that security at these sites would function better with less federal oversight. So starting this fall, the NRC will launch a pilot program allowing the power companies to design their own security exercises–a function formerly performed by federal terrorism experts. The industry says the new program will cost the plants less, yet allow for more frequent tests. But opponents, including many within the NRC, say the industry's track record has hardly earned it the right to looser regulation. In the past year alone, NRC inspectors have discovered alarms and video surveillance cameras that don't work, guards who can't operate their weapons, and guns that don't shoot. "I am very skeptical about the nuclear industry's ability to regulate itself," says Rep. Edward J. Markey, a vocal critic of nuclear security.

High on critics' lists of concerns is the failure rate in the NRC-run mock terrorist assaults–attacks that, if real, could have released radiation more lethal than the 1986 Chernobyl accident that resulted in an estimated 32,000 deaths. These exercises, called Operational Safeguards Response Evaluations, or OSREs, have been run by an outspoken former U.S. Navy SEAL captain named David Orrick. In a typical exercise, a team of three "terrorists" armed with small weapons and basic knowledge of how a plant works attempts to penetrate the facility. They evade or disable security equipment and destroy a set of targets in an effort to damage the plant's nuclear core, causing a radioactive release. In some cases, the mock terrorists make it all the way to the sensitive control room–even though they give plant operators ample advance notice of when they intend to strike.

Proponents of the NRC's mock attacks say they teach valuable lessons. In 1999, the Waterford 3 Nuclear Plant in Taft, La., failed a preliminary mock attack, but the plant's managers said that the exercise did not reflect the plant's true capability. So Orrick's team returned last year to conduct a more rigorous exercise against the plant. "We [the NRC team] just ate them alive," says one NRC inspector. The Waterford 3 site then hired more guards, improved training, and fortified physical barriers. They finally passed an NRC exercise last January. And in May, security guards easily apprehended a man with a history of mental illness who scaled a 10-foot, barbed-wire fence surrounding the site.

Still, critics charge that even the NRC's mock terrorist attacks do not reflect today's real-world scenarios. "There is nothing about protecting against a helicopter assault or a missile taking out one of our positions," says one plant security guard. Last September, for instance, an anti-nuclear demonstrator landed a motorized parafoil on the roof of a nuclear reactor in Bern, Switzerland, before being apprehended by security guards.

While nuclear plant operators design much of their security to prevent attacks from the outside, the record suggests that the greater danger lies within. "If somebody got a job as a janitor and got access to the plant, that's the real threat," says Erik Pakieser, former nuclear security officer at the Prairie Island nuclear generating plant in Minnesota. For instance, at the same time Donald Beauregard was cooking up his Project Worst Nightmare, a maintenance technician at the Crystal River site discovered that someone had intentional- ly disabled one of the plant's |emergency diesel generators. Some nuclear security experts also believe that sabotage should not have been ruled out so quickly as a possible cause of the 1979 accident at the Three Mile Island nuclear plant. Scientists at the Los Alamos National Laboratory found striking similarities between the incident and a computer-generated sabotage scenario they had run several months earlier.

Two decades later, critics remain troubled by the sorts of individuals who can gain access to a nuclear plant. In the early 1990s, a carpenter named Carl Drega got jobs at three nuclear power plants in the Northeast despite an arrest record and a job reference that described him as "volatile." Two months after Drega left the third plant, in 1997, he shot four people to death, including two state troopers, a judge, and a newspaper editor. An NRC investigation of the incident found that none of the three plants had violated their regulations by hiring him.

Easy access. Another insider, a computer programmer who once worked in the control room at the Maine Yankee nuclear power plant, goes to trial next year for murdering seven of his coworkers at a small Massachusetts technology company. Plant coworkers said the programmer, Michael McDermott, slept in a coffin and told a colleague he was sometimes so angry he felt like killing someone. In 1998, a worker at the Turkey Point nuclear plant in Florida had free access to critical areas of the plant for more than a month before officials learned of his 14 arrests. And at the Calvert Cliffs plant in Maryland, officials took eight months to learn that a worker was an illegal Mexican immigrant with fake identification papers and an arrest record. "Charles Manson could get access to a nuclear power plant," says former nuclear security officer Richard Kester.

But some experts worry that attackers can succeed even without getting inside. Classified reports from Sandia National Laboratories show that a well-placed truck bomb would not even have to enter a site's property to destroy vital equipment, leading to a possible release of radiation. In addition, experts say, the water-intake systems at some plants are particularly vulnerable to sabotage by either cutting off the water supply by clogging the intake valve or introducing volatile chemicals into the reactor's cooling system.

An even more accessible target may be spent nuclear material piling up at these plants. Large cooling pools inside reactor containment buildings were designed to store this fuel, but several years ago the pools began to fill up. Now, at many plants, the highly radioactive fuel is stored in cooling pools outside the containment building. "A lot of the spent nuclear fuel casks can be hit with a shoulder-fired missile by someone standing outside the fence," says Dave Lochbaum, nuclear safety engineer at the Union of Concerned Scientists. Yet at plants that are being decommissioned, the nuclear fuel is even less closely

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guarded. The Maine Yankee plant, which has stored 700 tons of spent fuel in outside cooling pools, has removed all of its vehi- cle barriers and received the NRC's permission to eliminate its armed guard force once the fuel is placed into dry casks.

The chairman of the NRC, Richard Meserve, says that no matter who runs the security drills, the plants remain among the world's most heavily guarded sites. And he says that the NRC mock attacks are expensive for both the commission to run and the plants to prepare for. "The reason we are making a big deal about this," says the Nuclear Energy Institute's Hendricks, is that the corrective actions resulting from these exercises " can have a tremendous impact" on a plant owner. "It can cost a million dollars to make these upgrades [of plant security]," she says. In any case, says Meserve, the new self-assessment pro- gram is only a trial: If it doesn't work, he says, it will be scrapped.

But the chorus of nuclear industry critics continues to grow. "The overall focus [at these sites] is not to protect the public but to get the NRC's blessing and ensure profits," says one nuclear security officer. Starting next week, the Waterford 3 plant, which had boosted security to pass the NRC's terrorist exercise, will begin to reduce its training programs and its guard force. "As soon as the NRC leaves," says one guard, "they downgrade security."

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Commencement Statistics

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Speeches

Paul Leventhal
Amanda Rader

Press Releases

NCI Chief Gives Commencement Address; Tetley and White also Honored

Amanda Rader is Williamson Medalist

The Late John Andrew Honored with Dewey Award

Conrad Kasperson Earns Lindback Award

Farrell, Lehman and Tyndall Earn Socrates Medals

Citations

Leventhal: Doctor of Laws

Tetley: Doctor of Fine Arts

White: Doctor of Science

Rader: Williamson Medalist

Andrew: Dewey Award

Kasperson: Lindback Award

Paul L. Leventhal, Class of '59
President, Nuclear Control Institute

Commencement Address to the Class of 2001
Franklin & Marshall College
Lancaster, PA
Sunday, May 13, 2001

It has been 42 years since I sat, as you are sitting, as a member of the graduating class of Franklin and Marshall College, looking up at the commencement speaker and thinking to myself, as you may well be thinking, "This guy is the last thing standing between me and my diploma."

With that in mind, let me tell you a story about Mario Cuomo, the former Governor of New York when he received his first invitation to speak at a graduation from his alma mater, St. John's University. He asked Father Flynn, the President of St. John's University how he should approach it. "Commencement speakers," said Father Flynn, "should think of themselves as the body at an old-fashioned Irish wake. They need you in order to have the party, but nobody expects you to say very much."

That's advice I intend to follow today, and I thank President Kneedler and the college trustees for giving me this opportunity to return to F & M and deliver my first graduation speech. And I will particularly cherish the Doctor of Laws degree I have just received. I am an ex-newspaperman who works in Washington and is always surrounded by lawyers. Now I outrank most of them.

At my graduation in 1959, I felt what many of you may be feeling today---above all, a magnificent and exultant sense of relief, but beneath it all, a persistent sense of confusion and trepidation about what lies ahead. Today marks your day of commencement, so let me help you put this glorious and conflicted moment of self esteem and self-doubt into perspective. Consider this the friendly shove that launches you on that proverbial journey of a thousand miles that begins with but a single step.

First, let me assure you that while that first step is a big one, you come well-equipped for the journey. I have grown to appreciate the remarkable education I received here at F & M, and I think over time, each of you will come to your own appreciation of what it is about this college that serves you so well over a lifetime. For me, it is an appreciation of Franklin & Marshall as, above all, a teaching school. Here there is a faculty of scholars who love to teach, and to reach the students they teach, and the impact these teachers had upon me was profound.

From Professor Vanderzell, in constitutional law, I learned of the

Farrell: Socrates Medalist

Lehman: Socrates Medalist

Tyndall: Socrates Medalist

overriding importance of the test of reasonableness, not only in assessing the political orientation of the Supreme Court at various periods in its history, but in evaluating the policies of the Executive Branch and the initiatives of Congress. The F & M Department of Government in those days had but three professors, extraordinary teachers all. We government majors used to joke about them being Philosopher Kings, the ruling elite from Plato's "The Republic." I owe a special debt to those three professors---John Vanderzell, Sid Wise and Dick Schier---because they disciplined my mind, taught me the values of social justice, stimulated my appetite for public affairs and--perhaps most important---got me used to unreasonable demands and hard work.

Which brings me back to the voyage upon which you are about to embark. What kind of trip will it be? You are embarking in an auspicious year.

The year 2001, thanks to the combined genius of screenwriter Arthur C. Clarke and director Stanley Kubrick, has a special meaning associated with human discovery and evolution. But while art often imitates life, life rarely lives up to the expectations of art. Mankind has not yet colonized the moon, much less sent an expedition of human beings to Saturn. Yet, the inscrutable complexities of this remarkable motion picture, especially its soaring finale of re-birth, somehow seem to be reflected in a technology-driven self-confidence that characterizes the human spirit we find in America today.

Compare this vision of 2001 with George Orwell's vision of 1984. To me and my F & M classmates in 1959, Orwell provided a disturbingly plausible version of what the future might hold. We had just lived through the political nightmare of McCarthyism and we were still living under the threat of nuclear war with the Soviet Union, a threat that was to reach its crescendo in the Cuban missile crisis three years later. Fortunately, 1984 came and went without a nuclear war, nor with an Orwellian Big Brother and his Thought Police displacing American democracy.

Yet, the question of whether American democracy is becoming dangerously dysfunctional has been the subject of a lively debate at F & M this year, with one visiting scholar warning that self-government is devolving into what he called "a process of socialization in which a political class tells us how to live." Today there is a growing sense of powerlessness in the face of Big Government, Big Money and the combining of the two into Big Power over our daily lives. More and more Americans are tuning out the politicians and the political process and are lasering in on making a good living and enjoying the good life. This pursuit of happiness, of course, is as American as apple pie, not to mention the Bill of Rights.

But these science- and political-fiction visions of the future are pertinent to the journey you begin today because, in truth, we have no way of predicting what the future will bring, or what our own role in shaping that future might be. Einstein once said, "Knowledge of what is does not open the door directly to what should be." But of one thing we can be quite sure. Whether we ultimately experience a great spiritual rebirth on the wings of technology, as envisioned in 2001, the movie, or a subjugation of the human spirit, as foreseen by Orwell, or a "return of the Stone Age on the wings of science," as Winston Churchill described the nuclear peril in 1946, depends not on some cosmic roll of the dice, but on a personal commitment by you to engage in the decisions by your government on which your future depends.

Be vigilant because, even in these seemingly good and peaceful times, there are dangers that are not widely understood and sometimes not easily seen, both at home and abroad. If we fail to recognize and reverse them, they could threaten our very survival. One such danger that I have been addressing in my work is the spread of nuclear weapons to additional nations and possibly even to terrorist groups. But when I set out 42 years ago as you are doing today, I had no way of knowing I would wind up, of all things, a nuclear non-proliferator.

I knew exactly what I wanted to do when I graduated from F & M. I wanted to be a journalist, and spent a decade in pursuit of this dream, most of it at Newsday as an investigative and political reporter. Journalism is a wonderful career because you never know what you might be covering next or where your work will lead. A classmate of mine at Columbia Graduate School of Journalism who was also a night copyboy with me at The New York Times was Joseph Lelyveld, now the Executive Editor of the Times. He recently told me, "What I've liked about this business from the start is that you can't see your path to the grave as clearly as you can in the more respectable professions."

I eventually came to feel, after covering politicians at the local, state and national levels, that a journalist was confined to being an observer, a commentator. The real challenge, I decided, was to get inside government and try to make it work. My big break came when the press secretary's job opened up in the office of a brainy, gutsy and workaholic United States Senator, Jacob Javits of New York, and he hired me to come to Washington. Javits was a liberal Republican. (In those days, "liberal Republican" was not an oxymoron!) I joined him in turbulent times at the start of the Nixon-Agnew Administration when Washington was under siege by anti-Vietnam War protesters. What a political learning experience that was!

By the way, perhaps the best piece of advice I ever got came from Javits, and I'd like to pass it on to you now. It was some years later when he was suffering the ravages of Lou Gehrig's disease. This disease destroys the body but leaves the brain intact. Sitting in a wheelchair and hooked to a respirator, Javits bid farewell to a reunion of his Senate staff "I leave you with four words," he whispered, which made us take notice because this Senator was famous for not being able to say anything in less than four thousand. And he said: "Demand excellence, take risks." Brilliant, I thought. Those words surely summed up his distinguished career, which began in poverty on New York's Lower East Side. And I have sought to apply them to my own work. And I say to you, if you demand excellence of yourself and those who may someday work for you, and you are prepared to take risks, you will probably reach where you are trying to go.

It's when I later went to work for a Democratic Senator, Abraham Ribicoff of Connecticut, that I had my first encounter with atomic energy. As an aide to a government operations subcommittee chaired by Ribicoff, I was assigned to handle a bill that came from the Nixon White House to reorganize the Atomic Energy Commission. We needed to transform the AEC into an all-energy agency capable of responding to challenges posed by the first Arab oil embargo of 1973. So, by chance of a bill referral, I was introduced to atomic energy.

For me, it was a baptism in fire, with no advanced warning, and it changed my life. This was the first time atomic energy legislation was assigned to a Congressional committee other than the then all-powerful Joint Committee on Atomic Energy, which had close

ties with the nuclear industry and bureaucracy. But I saw a need to do things different from the way the Joint Committee wanted things done---in particular, to make a clean break between regulation and promotion of atomic energy. There was a need to eliminate conflicts of interest inside the AEC that compromised the safety and security of nuclear power plants. So, with support from Chairman Ribicoff, we overcame objections from the Joint Committee and crafted a new law that "fissioned" the AEC--split it into the present-day Nuclear Regulatory Commission and Department of Energy.

There were some eye-openers for me in the course of doing this work. I learned that plutonium, the essential material of nuclear weapons, is produced in civilian nuclear power plants as a byproduct of generating electricity with uranium fuel. The United States was planning to extract plutonium from the highly radioactive, self-protecting spent fuel of these plants. And we were going to give permission to other countries to remove plutonium from the uranium we exported to them for use in their plants. This atom bomb material was going to be "recycled" as fuel in civilian nuclear plants. But there was a big problem, and that was that the level of protection against thefts and diversions of civilian plutonium was far below protection of military plutonium. And there was another problem: Plutonium is so poisonous that if a speck of it the size of a pollen grain gets caught in the lung, it causes cancer.

Through domestic law and regulation, we stopped the plutonium business in the United States. Spent fuel from U.S. nuclear power plants is now supposed to be disposed of as waste inside a mountain in Nevada, without recovering plutonium. But export controls enacted in another law I worked on--the Nuclear Non-Proliferation Act of 1978--failed to stop extraction of plutonium from fuel the United States supplied to Europe and Japan. And the flow of nuclear technology and materials from industrial countries to the developing world has continued. As a result, there is now more plutonium in civilian hands than in all of the nuclear weapons in the world. And some of it has already been turned into bombs, as in India, Pakistan and North Korea, while others have used or are now using civilian nuclear programs as a cover for weapons programs. Of greatest immediate concern are Iran and Iraq, and Japan's neighbors are wondering why the Japanese are accumulating so much plutonium.

I founded the Nuclear Control Institute 20 years ago after the Reagan landslide cost me my Democratic Senate job, then with Gary Hart of Colorado. I had just finished codirecting the Senate's investigation of the Three Mile Island nuclear accident and became acutely aware of that ineffable combination of human fallibility and mechanical failure that makes nuclear plants vulnerable to accidents, and also sabotage. This institute now serves as a research and advocacy center where work can continue on reducing nuclear dangers, especially on preventing the further spread of nuclear weapons and raising effective barriers against nuclear terrorism.

So, what lessons can I share with you from my own journey since graduating from Franklin & Marshall College? The first is this: Be prepared to be surprised, just as I was surprised to become a nuclear specialist. You never know where your work will lead. If you have no clear career objective now, that's O.K. The important thing is to get a job, work hard at it, and see where it leads. If you know what you want to do, pursue that career passionately and see if it leads to where you want to go.

I ended up working in public affairs. You surely don't get rich laboring in the public interest, but it can be an enriching experience. At the same time, if you manage to be effective in taking on powerful political and industrial interests, the frustrations can run high and the going can get rough. For the large majority of you who will work in the private sector, you still have a responsibility to engage in public affairs. President Eisenhower, in his famous farewell, warned of "the acquisition of unwarranted influenceÉby the military-industrial complex." But he also declared that "only an alert and knowledgeable citizenry" can ensure "that security and liberty may prosper together."

Vigilance and tenacity are absolutely essential. In public affairs, it is distressing to find that there is a tendency for issues to come full circle, and to find yourself back where you started. I will offer a few examples from my field.

- Nearly 30 years after Congress established an independent nuclear regulator, there are complaints from Capitol Hill that the nuclear power industry is being crippled by over regulation. The agency is now being intimidated by budget cuts to be more compliant. It has begun a process of granting life extension to America's aging fleet of 104 power reactors even in the face of a rash of forced shut downs due to equipment failures caused by aging. It was a forced shut-down that triggered the Three Mile Island accident.

- In addition, the security guards at half the nuclear power plants in the United States have failed to repel mock terrorist attacks against safety systems designed to prevent a reactor meltdown. These are socalled "force-on-force" exercises supervised by the Nuclear Regulatory Commission. The NRC refuses to take enforcement action in response to the failures, and is in the process of weakening the rules of the game in response to industry complaints. Sabotage of nuclear power plants may be the greatest domestic vulnerability in the United States today. This is the time to strengthen, not weaken, nuclear regulation.

- Some 25 years after enactment of the Nuclear Non Proliferation Act, there is a push on Capitol Hill to lift sanctions against nuclear and military transfers to India and Pakistan despite their nuclear weapons tests of 1998. Both countries used civilian nuclear power programs as a cover for development of nuclear weapons. Other nations known to be or suspected of developing nuclear weapons, like Iraq, North Korea and Iran, will be watching closely to see if U.S. and international sanctions against proliferation are weakening.

- The 20-year ban on use of plutonium fuel in U.S. power reactors is now at risk. There is a U.S.-Russian plan to dispose of excess military plutonium from retired warheads by using it as fuel in power reactors in both countries, rather than dispose of it directly as waste. The plutonium fuel plan raises safety and security risks, especially in Russia. But the Bush Administration has just zeroed out funding for the alternative approach of combining the excess plutonium with highly radioactive waste for disposal in the mountain in Nevada along with civilian spent fuel. And now plutonium advocates on

Capitol Hill are even suggesting that the program for geological disposal of spent fuel was a mistake and want to emulate the highly uneconomic and extremely risky European and Japanese plutonium programs.

- It's been more than 20 years since construction began on a U.S. nuclear power plant, but the Bush Administration may announce next week a plan to encourage nuclear power plant construction in response to electricity shortages and global warming. This policy is flawed for three reasons. First, new nuclear plants could not be brought on line quickly enough to offset present shortages, which are caused primarily by lack of electrical transmission capacity, not production capacity. Second, these plants could not make a big dent in global warming because two-thirds of carbon-dioxide emissions, a major contributor to global warming, come from transport and other non-electric sources. Third, turning to the world stage, if carbon-free nuclear plants were used to replace coal plants worldwide, there would have to be a 10fold increase to 3,000 nuclear plants. That would reduce carbon emissions by only 20%, but plutonium commerce would expand enormously, to millions of kilograms a year.

To give you an idea of the weapons significance of millions of kilograms of plutonium, listen to Dr. Theodore Taylor, who was America's most creative atomic bomb designer in the 1950s and is a member of our Institute's board. "The bomb that destroyed Nagasaki," he said, "set off an instant of explosive energy equivalent to a pile of dynamite as big as the White House that was contained in a sphere of plutonium no bigger than a baseball." That was a first-generation atomic bomb, made with about 6 kilograms of plutonium, and it is a technological feat that is now within the grasp of radical states or terrorists, if they manage to get their hands on the plutonium.

Ultimately it comes down to a test of reasonableness. Is it reasonable to assume, over time, that millions of kilograms of plutonium can be sequestered down to the few kilograms needed for a bomb that can destroy a city? This question, in my view, must be answered before giving any further comfort to an industry that remains officially committed to using plutonium as a fuel---and surely before supporting an extension and expansion of that industry in response to electricity shortages and global warming.

Today, there is an historic opportunity to turn away from plutonium, by supporting development of nonnuclear energy strategies and by supporting nuclear arms control and disarmament. The question is, will we seize this opportunity, or will we squander it?

On our Institute's original Board was the late historian, Barbara Tuchman, who in her book The March of Folly gave a sobering description of a phenomenon, one repeated throughout recorded history, that drives nations to destruction. Folly, she wrote, was "pervasive persistence in a policy demonstrably unworkable or counterproductive." To qualify as folly, she said, it "must have been perceived as counter-productive in its own time, not merely by hindsight,É(and) a feasible alternative course of action must have been available."

But on this joyous day, let me close on an optimistic note. I would not be in the business I am in if I were not an optimist. I remain

confident that our great nation will avoid the march of folly and steer the world clear of the slippery slope of nuclear proliferation. DeTocqueville, in observing democracy in a much younger America, wrote, "É(T)he great privilege enjoyed by the Americans is not only to be more enlightened than the other nations, but also to have the chance to make mistakes that can be retrieved."

I believe what deTocqueville said remains true to this day, and that an informed and engaged citizenry---led by you, the F & M graduates of 2001---will ensure that an enlightened America will endure and prevail.

Finally, let me say that I could not be in this business without the support and advice of my wife, Sharon Tanzer, the vicepresident of Nuclear Control Institute, who is here today with our sons, Ted and Josh, and without the further encouragement of my brother, Warren Leventhal, F & M Class of '53, and his wife Gloria, who are also here today.

To the Class of 2001, my congratulations to you all, for all the hard work that has brought you to this day of commencement--and for all your achievements in days to come. And on this Mother's Day, congratulations to the mothers, and the fathers, too, whom you have made so very proud.

Thank you and good luck to you all.

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Career TRACKS

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In the news: Indian Point 2

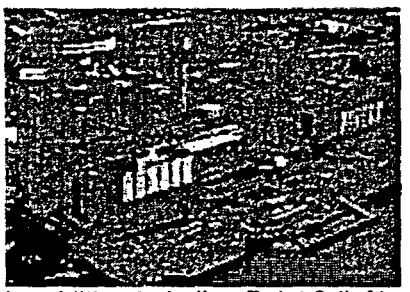
On Feb. 15, 2000, ConEdison's Indian Point 2 nuclear generating station in Buchanan sustained the most serious accident in its 26-year operating history.
This page includes key developments in the continuing story, as reported by The Journal News.

NRC flags ConEd with red tag for Indian Point

By SHAWN COHEN
The Journal News
(Original publication: 11/21/2000)

BUCHANAN — Indian Point 2 gained a dubious distinction yesterday when it became the first of the nation's 103 commercial nuclear power plants to be designated "red" — the highest risk assessment — by federal regulators for failing to detect flaws in a steam generator tube before the February radiation leak.

As a result of this violation, Consolidated Edison will face the Nuclear Regulatory Commission's highest level of scrutiny for at least a year. It will be required to open its doors to more frequent and thorough inspections, as well as pay for them. The NRC all but blamed Con Edison for the tube rupture, saying the utility should have detected corrosion during a 1997 inspection.



In addition to Indian Point 2 (left) the nuclear complex in Buchanan includes defunct Indian Point 1 (low dome in center) and Indian Point 3.

ConEd knew steam generators were defective

For more than 20 years, ConEd operated steam generators that it knew were defective. In fact, ConEd and other utilities had sued the manufacturer of the steam generators, and when the case was settled ConEd was given replacements.

But ConEd did not install the replacements until after a coolant pipe rupturned releasing radioactive water and steam. (Story published 2/18/2001)

"The (NRC) team concluded these failures resulted in tubes with flaws being left in service following the 1997 inspection until one of these tubes failed in February," the NRC said, adding that this "resulted in a significant reduction in safety margin during the plant's operating cycle."

The Feb. 15 emergency began when a steam generator tube cracked and began spewing radioactive water at a rate of 100 gallons a minute inside the plant. Con Edison declared the plant's first-ever emergency alert and quickly brought the leak under control. The plant, which is replacing its steam generators and hopes to reopen next month, has come under greater scrutiny since February.

During the spring, the NRC added Indian Point 2 to its "agency focus" list as part of a process designed to help the NRC keep an eye on troublesome plants. The utility is one of just two plants on the list. At the same time, the NRC established the color codes, ranging in severity from green to white to yellow to red, to assess specific plant problems.

In a preliminary report in August, the NRC categorized the 1997 inspection as a "red" violation of "high safety significance." It criticized Con Edison's testing methods, saying it could have noticed flaws, had it used an electrical probe properly and could have prevented the leak had it

Timeline

This timeline lists key events since the Indian Point accident, as reported in The Journal News on the dates shown. Click on any date to read a news story from our archives. Stories are listed chronologically; for the latest report, start at the of the timeline.

Feb. 16, 2000
For the first time in the 26-year history of the plant, ConEd declares an emergency at Indian Point 2 after a coolant pipe ruptures and spills radiocative water.

Feb. 17, 2000
The state of emergency is lifted and ConEd begins investigating the cause of the accident.

Feb. 19, 2000
A team of seven NRC inspectors are en route to Buchanan for an on-site review of ConEd's response to the accident.

Feb. 23, 2000
Westchester County officials voice skepticism about ConEd's preparedness for a nuclear accident.

March 1, 2000
Small amounts of radiation may have leaked into the air during the Feb. 15 incident, but the NRC said the levels were too low to be detected.

March 6, 2000
Ten years ago, ConEd said it would consider replacing the steam generators. **March 15, 2000**
ConEd tells the NRC that it plans to plug the leaky pipe and make other repairs, but won't replace the steam generator.

March 17, 2000
The NRC says the crack that

taken corrective actions.

Con Edison, which challenged this in September when it urged a "yellow" finding, raised objections again yesterday.

"As we have consistently said since February, our 1997 steam generator inspections used the most sophisticated technology and analysis at the time," said Con Edison spokesman Michael Clendenin.

"We submitted all required inspection documentation, which was reviewed and subsequently approved by the NRC. We believe all of our actions were thorough and prudent," Clendenin said.

But yesterday, the NRC stood firm. The red finding lasts a year and could be extended if Con Edison doesn't show signs of improvement, said Neil Sheehan, spokesman for the NRC Region 1.

"If they get several red findings against them, they could end up in a situation where we might shut them down," Sheehan said.

He noted that Con Edison received a "white" violation for its handling of a reactor shutdown in August 1999, during which an emergency diesel generator malfunctioned.

The NRC's inspector general has said the NRC itself was partly responsible for the 1997 inspection, because it failed to catch mistakes in the data. But the caused the February leak might have been visible during a 1997 inspection of the plant.

March 25, 2000
Modifying an earlier report, the NRC says flaws discovered in a 1997 inspection could not have predicted the radiocative leak.

March 30, 2000
Federal regulators say ConEd responded properly and promptly to the emergency.

April 8, 2000
Public interest groups and nuclear watchdog organizations urge the NRC to require ConEd to replace the steam generators, not repair them.

May 2, 2000
ConEdison impurities in the water for causing the corrosion that led to the leak in the steam generator.

June 3, 2000
ConEd seeks permission to restart the plant with repaired steam generators, while promising to replace them by year-end.

June 7, 2000
The NRC's inspector general reveals an investigation into an agency decision to allow ConEd to forego a scheduled 1999 inspection of the steam generator.

June 21, 2000
An internal memo prepared by ConEd and obtained by The Journal News shows that the power plant was in turmoil in the hours following the accident.

Aug. 10, 2000
ConEdison scraps plans to repair the Indian Point steam generators, but will replace them instead.

Sept. 1, 2000

agency has placed sole blame on Con Edison.

Con Edison has announced it will sell its Indian Point 1 and 2 plants for $502 million to New Orleans-based Entergy Corp.

An inspector general's report says the NRC could have prevented the accident by more carefully scrutinizing a 1997 inspection report.

Sept. 1, 2000

Although President Clinton's new home is only 12 miles away, the Secret Service was not alerted about the incident.

Sept. 12, 2000

A key ConEd manager says "unwarranted pride" among plant operators contributed to the February accident.

Oct. 12, 2000

The NRC rejects a petition from nuclear safety advocates that would have made it more difficult for ConEd to reopen Indian Point 2.

Oct. 12, 2000

The NRC orders closer scrutiny of Indian Point 2.

ConEd's Indian Point 2 becomes the first of the nation's 103 private nuclear power plants to be adorned with a "red" tag -- the NRC's highest risk assessment.

Nov. 22, 2000

Groups that had been critical of ConEd and also criticized the NRC are praising the regulatory commission for its decision to red-tag the utility in the ongoing response to the February leak.

Dec. 11, 2000

Con Edison says it has completed repairs to the Indian Point 2 power plant, and is days away from restarting the reactor and generating electricity.

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Figures From The CRAC-2 Report

"Calculation of Reactor Accident Consequences"

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a 1982 study by Sandia Labs, New Mexico

commissioned by the US Nuclear Regulatory Commission

The following figures are from the CRAC-2 Report.

This report looks at nuclear power plants in the USA and assumes a class 9 [worse case] meltdown

The report was commissioned by the NRC [Nuclear Regulatory Commisssion] and carried out by Sandia Labs in New Mexico.

It was published by the U.S. Congress and the Washington Post on November 1, 1982.

Name of Power Plant	Location of Plant	Peak Early Fatalities	Peak Early Injuries	Peak Deaths from Cancer	Property Damage
Arkansas Nuclear Unit One	Russelville, Arkansas	1,190	3,400	2,900	$68.1 billion

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Arkansas Nuclear Unit Two		2,100	4,000	3,000	$84.9 billion
Beaver Valley Unit One	Shippingport, Pennsylvania	19,000	156,000	28,600	$122 billion
Beaver Valley Unit Two		19,000	156,000	24,000	$11 billion
Bellefonte Unit One	Scotsboro, Alabama	3,600	7,700	4,500	$86.1 billion
Bellefonte Unit Two		3,600	7,700	4,500	$82.7 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Big Rock Point Unit One	Charlevoix, Michigan	3,450	6,030	10,900	unavailable
Braidwood Unit One	Joliet, Illinois	6,750	63,300	14,200	$127 billion
Braidwood Unit Two		6,750	63,300	14,200	$122 billion
Brown's Ferry Unit One	Decatur, Alabama	18,000	42,000	3,800	$67.3 billion
Brown's Ferry Unit Two		18,000	42,000	3,800	$69.1 billion
Brown's Ferry Unit Three		18,000	42,000	3,800	$73.0 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Brunswick Unit One	Brunswick, N. Carolina	7,500	18,000	4,600	$56.5 billion
Brunswick Unit Two		7,500	18,000	4,600	$53.9 billion

Byron Unit One	Rockford, Illinois	9,050	79,300	15,300	$114 billion
Byron Unit Two		9,050	79,300	15,300	$114 billion
Callaway Unit One	Callaway, Missouri	11,500	32,000	9,600	$110 billion
Calvert Cliffs Unit One	Lusby, Maryland	5,600	15,000	23,000	$87.4 billion
Calvert Cliffs Unit Two		5,600	15,000	23,000	$92.0 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Catawba Unit One	Rock Hill, S. Carolina	42,000	88,000	5,800	$101 billion
Catawba Unit Two		42,000	88,000	5,800	$93.7 billion
Clinton Unit One	Clinton, Illinois	1,600	32,000	13,000	$92.8 billion
Commanche Peak Unit One	Glen Rose, Texas	1,200	14,000	4,800	$117 billion
Cook Unit One	Bridgman, Illinois	1,900	80,000	13,000	$91.9 billion
Cook Unit Two		2,000	88,000	13,000	$101 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Cooper Unit One	Brownsville, Nebraska	1,600	2,800	3,100	$57.2 billion
Crystal River Unit Three	Red Level, Florida	900	3,800	2,800	$53.8 billion
Davis-Besse Unit Three	Ottawa, Ohio	1,400	73,000	10,000	$84.0 billion

Diablo Canyon Unit One	San Luis Obispo, California	10,000	11,000	12,000	$155.0 billion
Diablo Canyon Unit Two		10,000	12,000	12,000	$158.0 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Dresden Unit One	Morris, Illinois	6,000	4,000	5,000	$23.5 billion
Dresden Unit Two		42,000	39,000	13,000	$87.4 billion
Dresden Unit Three		42,000	39,000	13,000	$89.6 billion
Duane Arnold Unit One	Palo, Indiana	scaled 2,900	12,000	not visible	$53.8 billion
Farley Unit One	Dothan, Alabama	12,000	12,000	2,900	$52.2 billion
Farley Unit Two		12,000	12,000	2,900	$59.1 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Fermi Unit Two	Laguna Beach, Michigan	8,000	340,000	13,000	$136.0 billion
Fitzpatrick Unit One	Scriba, New York	scaled 1,000	16,000	17,000	$103.0 billion
Fort Calhoun Unit One	Washington, Nebraska	Scaled 3,000	32,000	3,000	$43.5 billion
Fort St. Vrain Unit One	Fort St. Vrain, Colorado	Scaled 3,000	3,000	1,000	$38.8 billion
Ginna Unit One	Ontario, New York	Scaled 2,000	28,000	14,000	$63.0 billion
Grand Gulf Unit One	Vicksburg, Massachusetts	4,500	10,000	3,800	$83.0 billion

Grand Gulf Unit Two		4,500	10,000	3,800	$70.7 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Haddam Neck Unit One	Haddam Neck, Connecticut	Scaled 29,000	50,000	23,000	$74.1 billion
Hatch Unit One	Baxley, Georgia	700	4,000	3,000	$51.0 billion
Hatch Unit Two		700	4,000	3,000	$56.0 billion
Indian Point Unit Two	Buchanon, New York	46,000	141,000	13,000	$274.0 billion
Indian Point Unit Three		50,000	167,000	14,000	$314.0 billion
Kewaunee Unit One	Carlton, Wisconsin	Scaled 900	17,000	8,000	$46.9 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Lasalle Unit One	Ottawa, Illinois	14,000	12,000	15,000	$118.0 billion
Lasalle Unit Two		14,000	12,000	15,000	$120.0 billion
La Crosse Unit One	La Crosse, Wisconsin	Scaled 70	400	200	$16.0 billion
Limerick Unit One	Montgomery, Pennsylvania	74,000	610,000	34,000	$213.0 billion
Limerick Unit Two		74,000	610,000	34,000	$197.0 billion
Maine Yankee Unit One	Wiscasset, Maine	Scaled 8,000	10,000	21,000	$76.5 billion

Name of Power Plant	Location of Plant	Peak Early Fatalities	Peak Early Injuries	Peak Deaths from Cancer	Property Damage
Marble Hill Unit One	Jefferson, Indiana	12,000	150,000	8,000	$87.2 billion
Marble Hill Unit Two		12,000	150,000	8,000	$83.8 billion
McGuire Unit One	Cornelius, N. Carolina	12,000	21,000	26,000	$106.0 billion
McGuire Unit Two		12,000	21,000	26,000	$110.0 billion
Midland Unit One	Midland, Michigan	4,000	11,000	8,000	$56.1 billion
Midland Unit Two		7,000	20,000	10,000	$80.4 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Millstone Unit One	Waterford, Connecticut	13,000	12,000	28,000	$91.5 billion
Millstone Unit Two		18,000	18,000	33,000	$135.0 billion
Millstone Unit Three		23,000	30,000	38,000	$174.0 billion
Monticello Unit One	Monticello, Minnesota	Scaled 500	10,000	4,000	$44.6 billion
Nine Mile Point Unit One	Scriba, New York	800	11,000	14,000	$66.2 billion
Nine Mile Point Unit Two		1,400	26,000	20,000	$134.0 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**

North Anna Unit One	Mineral, Virginia	1,800 [5,400 to 7,200]	5,000	29,000	$66.0 billion
North Anna Unit Two		1,800 [5,400 to 7,200]	5,000	29,000	$60.3 billion
Oconee Unit One	Seneca, S. Carolina	4,000 [1,200 to 16,000]	47,000	4,000	$56.8 billion
Oconee Unit Two		4,000 [1,200 to 16,000]	47,000	4,000	$58.3 billion
Oconee Unit Three		4,000 [1,200 to 16,000]	47,000	4,000	$58.3 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Oyster Creek Unit Three	Toms River, New Jersey	13,000 [39,000 to 52,000]	10,000	23,000	$79.8 billion
Palisades Unit Three	South Haven, Michigan	Scaled 1,000 [3,000 to 4,000]	7,000	Scaled 10,000	$52.6 billion [1982 dollars]
Palo Verde Unit One	Maricopa, Arizona	4,000 [12 to 16,000]	36,000	15,000	$89.7 billion
Palo Verde Unit Two		4,000 [12 to 16,000]	36,000	15,000	$89.7 billion
Palo Verde Unit Three		4,000 [12 to 16,000]	36,000	15,000	$89.7 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Peach Bottom Unit Two	Peach Bottom, Pennsylvania	72,000 [216,000 to 288,000]	45,000	37,000 [111,000 to 148,000]	$119.0 billion

St. Lucie Unit Two		5,000 [15 to 20,000]	6,000	3,000	$59.1 billion
Salem Unit One	Salem, New Jersey	100,000 [300,000 to 400,000]	70,000	40,000	$135.0 billion
Salem Unit Two		100,000 [300,000 to 400,000]	70,000	40,000	$150.0 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
San Onofre Unit One	San Clemente, California	8,000 [24 to 32,000]	6,000	10,000	$58.8 billion
San Onofre Unit Two		27,000 [81,000 to 108,000]	23,000	18,000	$186.0 billion
San Onofre Unit Three		27,000 [81,000 to 108,000]	23,000	18,000	$182.0 billion
Seabrook Unit One	Seabrook, New Hampshire	7,000 [21 to 28,000]	27,000	6,000	$163.0 billion
Seabrook Unit Two		7,000 [21 to 28,000]	27,000	6,000	$150.0 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Sequoyah Unit One	Daisy, Tennessee	29,000 [87,000 to 116,000]	61,000	4,700	$96.8 billion
Sequoyah Unit Two		29,000 [87,000 to 116,000]	61,000	4,700	$98.6 billion
Shearon Harris Unit One	Apex, N. Carolina	11,000 [33,000 to 44,000]	31,000	6,000	$68.5 billion
Shearon Harris Unit Two		11,000 [33,000 to 44,000]	31,000	6,000	$47.8 billion

Peach Bottom Unit Three		72,000 [216,000 to 288,000]	45,000	37,000 [111,000 to 148,000]	$119.0 billion
Perry Unit One	Painesville, Ohio	5,500 [16,500 to 22,000]	180,000	14,000	$102.0 billion
Perry Unit Two		5,500 [16,500 to 22,000]	180,000	14,000	$86.8 billion
Pilgrim Unit One	Plymouth, Massachusetts	3,000 [9,000 to 12,000]	30,000	23,000	$81.8 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Point Beach Unit One	Two Creeks, Wisconsin	500 [1,500 to 2,000]	9,000	7,000	$41.4 billion
Point Beach Unit Two		500 [1,500 to 2,000]	9,000	7,000	$41.4 billion
Prairie Island Unit One	Red Wing, Minnesota	2,000 [6 to 8,000]	4,000	4,000	$48.3 billion
Prairie Island Unit Two		2,000 [6 to 8,000]	4,000	4,000	$49.5 billion
Quad Cities Unit One	Cordova, Illinois	12,000 [36 to 48,000]	41,000	12,000	$65.1 billion
Quad Cities Unit Two		12,000 [36 to 48,000]	41,000	12,000	$65.1 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Rancho Seco Unit One	Clay Station, California	Scaled 30,000 [90 to 120,000]	34,000	Scaled 6,000	$113.0 billion
Robinson Unit One	Hartsville, S. Carolina	Scaled 2,000 [6 to 8,000]	8,000	3,000	$42.5 billion
St. Lucie Unit One	Fort Pierce, Florida	5,000 [15 to 20,000]	6,000	3,000	$54.3 billion

Shoreham Unit One	Wading River, New York	Scaled 40,000 [120,000 to 160,000]	75,000	35,000	$157.0 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
South Texas Unit One	South Texas, Texas	18,000 [54,000 to 72,000]	10,000	4,000	$112.0 billion
South Texas Unit Two		18,000 [54,000 to 72,000]	10,000	4,000	$104.0 billion
Summer Unit One	Fairfield, S. Carolina	5,000 [15 to 20,000]	73,000	4,000	$68.2 billion
Surry Unit One	Gravel Neck, Virginia	31,000 [93,000 to 124,000]	36,000	23,000	$56.3 billion
Surry Unit Two		31,000 [93,000 to 124,000]	36,000	23,000	$57.8 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Susquehanna Unit One	Berwick, Pennsylvania	67,000 [201,000 to 268,000]	47,000	28,000	$143.0 billion
Susquehanna Unit Two		67,000 [201,000 to 268,000]	47,000	28,000	$137.0 billion
Three Mile Island Unit One	Middletown, Pennsylvania	42,000 [126,000 to 168,000]	50,000	26,000	$102.0 billion
Three Mile Island Unit Two		46,000 [138,000 to 184,000]	57,000	28,000	$122.0 billion
Trojan Unit One	Prescott, Oregon	Scaled 1,000 [3 to 4,000]	14,000	Scaled 5,000	$89.7 billion

Name of Power Plant	Location of Plant	Peak Early Fatalities	Peak Early Injuries	Peak Deaths from Cancer	Property Damage
Turkey Point Unit Three	Florida City, Florida	29,000 [87,000 to 116,000]	45,000	4,000	$43.6 billion
Turkey Point Unit Four		29,000 [87,000 to 116,000]	45,000	4,000	$48.6 billion
Vermont Yankee Unit One	Vernon, Vermont	7,000 [21 to 28,000]	3,000	17,000	$68.8 billion
Vogtle Unit One	Burke, Georgia	200 [6 to 800]	39,000	4,000	$70.3 billion
Vogtle Unit Two		200 [6 to 800]	39,000	4,000	$62.3 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**
Waterford Unit Three	St. Charles, Louisiana	96,000 [288,000 to 384,000]	279,000	9,000	$131.0 billion
Watts Bar Unit One	Rhea, Tennessee	5,000 [15 to 20,000]	11,000	4,000	$86.6 billion
Watts Bar Unit Two		5,000 [15 to 20,000]	11,000	4,000	$83.3 billion
WPPSS Unit One	Richland, Washington	19,000	156,000	4,000	$80.4 billion
WPPSS Unit Two	Benton, Washington	300 [900 to 1,200]	17,000	4,000	$77.3 billion
WPPSS Unit Three	Olympia, Washington	173 [519 to 492]	13,800	4,000	$73.7 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**

Wolf Creek Unit One	Burlington, Kansas	1,000 [3 to 4,000]	3,000	3,000	$105.0 billion
Yankee Rowe Unit Three	Rowe, Massachusetts	1,000 [3 to 4,000]	100	4,000	$21.4 billion
Zimmer Unit Three	Moscow, Ohio	9,000 [27 to 36,000]	109,000	10,000	$84.5 billion
Zion Unit One	Zion, Illinois	14,000 [42 to 56,000]	155,000	17,000	$146.0 billion
Zion Unit Two		14,000 [42 to 56,000]	155,000	17,000	$146.0 billion
Name of Power Plant	**Location of Plant**	**Peak Early Fatalities**	**Peak Early Injuries**	**Peak Deaths from Cancer**	**Property Damage**

[Accident Possibilities at Gentilly-2]

[Findings on CANDU Safety]

[Reactor Accidents Sub-Directory] [COMPLETE DIRECTORY]

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The New York Times, May 6, 1983

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The New York Times

May 6, 1983, Friday, Late City Final Edition

SECTION: Section B; Page 5, Column 1; Metropolitan Desk

LENGTH: 2335 words

HEADLINE: AT INDIAN PT., A HISTORY OF NUCLEAR **POWER,** PROBLEMS AND CONTROVERSY

BODY:
More than 30 years ago, in 1952, the Consolidated Edison Company began to study the possibility of using atomic energy to generate electricity. It became the first company to announce plans to build a commercial-sized atomic **power plant** without government subsidy.

Oct. 8, 1954. Con Edison announces the purchase of a 250-acre site in Buchanan, N.Y., on the east bank of the Hudson River 35 miles north of midtown Manhattan, to construct its atomic **power plant.** It says the **plant,** to be built in an area known as Indian Point, will cost $50 million to build.

May 4, 1956. The Atomic Energy Commission, the predecessor of today's Nuclear Regulatory Commission, grants Con Edison a construction permit. Major work on the **plant,** whose cost has risen to $121 million, starts in 1958. The **plant** - with a capacity of 275 megawatts, 59 percent from the heat of the reactor and 41 percent from a supplementary oil-fired superheater - is expected to provide enough **power** to fill the needs of a million homes.

Feb. 23, 1962. The Atomic Energy Commission, after finding that Con Edison has provided "reasonable assurance that the reactor could be operated so as not to endanger the health and safety of the public," authorizes a provisional 18-month license for its operation. The **plant,** still under construction, is designed to produce 1.79 billion kilowatt-hours of electrical energy a year at an operating cost of $25 million.

Sept. 16, 1962. The reactor, which would later become known as Indian Point 1, is to begin generating **power** today and and go into commercial operation next month. Its generating capacity is about a quarter of that of the average reactor that would be built two decades later.

Dec. 10, 1962. Con Edison applies for a permit to build another reactor, in Ravenswood, Queens. After public protests, the application is withdrawn on Jan. 6, 1964.

May 27, 1965. Seven United States Representatives from New York State accuse state officials of covering up fish kills in the Hudson River near the Indian Point **power plant.**

Nov. 23, 1965. Con Edison's directors approve plans to build a second nuclear reactor at Indian Point. Cost of construction is placed at $108 million. Gov. Nelson A. Rockefeller calls the move "of major importance to our state and its expanding atomic industry."

April 1967. Con Edison applies for permission to build a third nuclear **plant** at Indian Point. Permission is granted in August 1969, and construction begins shortly afterward.

May 13, 1970. The state charges Con Edison with serious violations of state conservation laws in the operation of its nuclear generating **plant** and asks that the **plant** be closed until "suitable methods" to protect the Hudson River can be developed. State seeks $5 million in damages for fish kills resulting from **plant's** operation.

June 30, 1970. Indian Point 1 is shut down because of defects in the stainless steel piping used to help keep the reactor cool. Meanwhile, the fish kills, caused by hot water being discharged into the Hudson River from the **plant's** cooling systems, have drawn protests from fishermen and have resulted in a $1.6 million fine from the state.

May 1972. The state levies $1.5 million in fines against Con Edison for a "massive" killing of fish in the Hudson River. The total fine is based on a civil penalty of $500 plus $10 for each fish killed.

Oct. 20, 1972. Con Edison withdraws an application to build nuclear **power** generating **plants** in Verplanck, near the Indian Point **plants.** One month later, the utility begins looking for sites away from the Hudson River to build generating **plants** after increasing pressures from environmentalists and others to preserve the Hudson River environment.

June 26, 1973. Indian Point 2 - which generates 873 megawatts, enough energy for almost 400,000 homes - goes into operation. September 1973. The commission rules that Con Edison must construct a "closed cycle" cooling system at its Indian Point **plants** that would prevent the release of the hot water into the Hudson and, therefore, avoid destruction of fish breeding grounds.

Dec. 1, 1973. Con Edison acknowledges that its new **power plant,** Indian Point 2, has significant problems. The acknowledgement comes after an accident on Nov. 13 forced it to shut down. Con Edison officials says the problems centered around a buckling and bulging of the steel liner in the reinforced concrete dome in which the nuclear **plant** is housed. At the time of the announcement, the other Con Edison nuclear **plant,** Indian Point 1, has been inoperable for more than a year.

Feb. 28, 1974. The State **Power** Authority announces plans to build a nuclear **power plant** along the Hudson River at either Athens or Cemeton, in Greene County, in 1982. The **plant,** along with another proposed, are needed, according to the state, to supply energy needs of the Metropolitan Transportation Authority. The **plants** are never built.

Sept. 18, 1974. The State Legislature votes to allow the **Power** Authority of the State of New York to take over Con Edison's Indian Point 3 **plant,** which is still under construction.

Oct. 14, 1974. Indian Point 1 is permanently shut down, 12 years after going into operation, because it lacks an emergency corecooling system, which the Nuclear Regulator Commission insists be added for continued operation. Techncially, the **plant** is retired but not decomissioned.

Dec. 30, 1975. Under pressure from Gov. Hugh Carey, Con Edison agrees to sell Indian Point 3 to the State **Power** Authority. The action disappoints environmentalists who hoped the sale could be blocked and eventually the **plant** closed down as a threat to the Hudson River and the adjacent Westchester County area.

Jan. 20, 1976. William N. Anders, chairman of the Nuclear Regulatory Commission, orders two separate investigations of two Indian Point reactors after **Robert** D. Pollard, the commission's

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project manager for Indian Point 3, questioned the safety of the **plants.** Mr. Pollard accused the commission of suppressing the existence of unresolved safety problems. On Jan. 13, Mr. Pollard resigned from the commission's staff, after four months on the job. After hearings in Washington, the N.R.C. decided to take no action.

Aug. 30, 1976. Indian Point 3 - with a capacity of 965 megawatts, 75 percent of which is used by government agencies, such as the Metropolitan Transportation Authority - goes into operation.

October 1979. An accident at the Three Mile Island **power plant** in Middletown, Pa., spurs a crescendo of criticism about the locations of several of the nation's 72 nuclear reactors and the emergency planning that has been conducted at virtually all of them. President Carter appoints appoints a commission to investigate the accident and to consider whether any of the nation's existing reactors should be shut down.

Nov. 18, 1979. **Robert Ryan,** the director of the Nuclear Regulatory Commission's Office of State Programs, testifies before the President's Commission on the Accident at Three Mile Island. He says the area around the Indian Point **plants** is one of the most populous of any containing nuclear **plants** and calls their construction there "**insane.**"

Dec. 18, 1979. The N.R.C. says that emergency evacuation plans for Indian Point are "lacking" "lacking" in key areas. The commission gives Con Edison and the State **Power** Authority two months to submit revisions.

Jan. 29, 1980. The two nuclear **plants** at Indian Point have been shut down for five months for refueling, maintenance and repairs, and the N.R.C. says safety improvements must be completed before they can resume operation. Eventually, Indian Point 2 resumes operation in early February and Indian Point 3 returns to service on Feb. 16.

Feb. 27, 1980. Con Edison accedes to the wishes of the N.R.C. and retires Indian Point 1 permanently. It had been inoperable since 1974. "It's the end of the beginning," says William J. Cahill, Jr., a Con Edison vice-president.

Oct. 17, 1980. A major water leak in a joint of a pipe carrying nonradioactive water deposits about 100,000 gallons of water in a containment building of Indian Point 2.

Oct. 24, 1980. The N.R.C. orders Indian Point 2 shut down until Con Edison determines how the leak went unnoticed. Five days later, the commission initiates an investigation to determine why Con Edison failed to notify the commission about the leak.

Dec. 10, 1980. The N.R.C. fines Con Edison $210,000 for the flooding that occurred at its Indian Point **plant** in October.

Jan. 7, 1981. The staff of the Public Service Commission says Con Edison should be forced to refund $43 million to its customers because of a "substantial likelihood" that "imprudent actions" by the utility caused the water leak that forced the **plant** to shut down. The money, which is eventually refunded to customers, represented extra fuel charges that were passed on to customers by Con Edison when it was forced to buy more expensive **power** to replace what would have been generated by the disabled **plant.**

Jan. 31, 1981. The State **Power** Authority shuts down Indian Point 3 because of malfunction in the **plant's** steam turbine section.

Meanwhile, officials from the four-county area within 10 miles of Indian Point testify at hearings in White Plains that an emergency evacuation plan presented by the utilities contains major flaws and urges that they not be approved by the N.R.C.

April 7, 1981. The technical staff of the N.R.C. says the Indian Point 2 nuclear reactor apparently suffered no damage from the flooding accident last October. The decision clears the way for the unit to be restarted. But one month later, Con Edison announces it has run into still another mechanical problem with Indian Point 2 and will not be able to put **plant** back in service until the end of May.

May 26, 1981. After the longest shutdown in its eight-year history, Indian Point 2 resumes service.

Dec. 11, 1981. Con Edison is fined $40,000 by the N.R.C. for not protecting workers from radiation at its Indian Point 2 **plant.**

Dec. 19, 1981. During 1981, Indian Point 2 has operated an average of only 11 days a month, according to Con Edison.

Feb. 27, 1982. A report by the Rand Corporation concludes that the cost of closing the Indian Point nuclear **power plants** has consistenly been underestimated and may be as high as $17.4 billion. If the **plant** were closed, the report says, electricity prices in New York City and Westchester County would increase substantially almost immediately.

March 3, 1982. Indian Point simulates a major accident to test emergency evacuation plans. Federal officials will later say that the drill was "generally good," although there were areas that need strengthening.

March 12, 1982. A plan to protect the nearly 300,000 people who live near Indian Point in case of a nuclear accident is said to be inadequate and may never be workable, according to some officials at the county, town and school-district level who would be responsible for carrying it out.

March 25, 1982. A leak and corrosion of steam generator tubes at the Indian Point 3 **power plant** forces the **plant** to close, and as of May 1983, it is still closed.

Aug. 2, 1982. The Nuclear Regulatory Commission threatens to close Indian Point unless flaws in the emergency evacuation plans for the area surrounding the **plant** are corrected within the next four months. The commission cites deficiencies in provisions for notifying residents in the area, for educating the public in advance about what to do, for making agreementrs with bus companies to provide emergency service and for limiting exposure of emergency workers to radiation.

Sept. 2, 1982. The chairman of a three-judge panel conducting hearings on safety of the two Indian Point **plant** resigns, saying the N.R.C. was not giving opponents of the **plant** a fair chance to state their case. The chairman, Administrative Law Judge Louis J. Carter, had been presiding at hearings ordered by the commission.

Dec. 17, 1982. The Federal Emergency Management Agency, the federal agency charged with evaluating prepardeness for an accident at Indian Point, says the emergency **plants** were "not feasible" because of significant deficiencies. The agency says it will decide whether to fine the utility operators, suspend the **plant's** operating licenses or take other action.

Dec. 22, 1982. The N.R.C. votes 3 to 2 to permit the Indian Point nuclear reactors to operate and to wait until an accident drill in March to determine whether deficiencies in emergency planning had been corrected. The **plants** had been under a Dec. 3 deadline to correct deficiencies found after a drill last March.

Jan. 31, 1983. Residents living within 10 miles of Indian Point **plants** indicate in a poll that

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Jan. 31, 1983. Residents living within 10 miles of Indian Point **plants** indicate in a poll that

they are confused about the evacuation plan and unlikely to follow it once informed.

Feb. 19, 1983. All but one of 75 warning sirens in the area surrounding Indian Point function well in a test. In a drill in March 1982, sirens malfunctioned and residents reported that they never heard a sound.

March 1, 1983. Testifying before a House subcommittee, John S. Dyson, head of the State **Power** Authority, says opponents of nuclear **power plants** "tend to be zealots who don't care in a sense what the answers are."

March 9, 1983. From bus drivers to county executives to the Lieutenant Governor, 2,000 people test their ability to respond to a major accident at the Indian Point **plants.** Watching closely are 55 Federal inspectors aznd one of the five members of Nuclear Regulatory Commission.

April 15, 1983. 7The Federal Emergency Management Agency issues a report based on the March drill that concludes the area around Indian Point is not prepared for a possible accident at the nuclear reactors, and the safety for the 288,000 people living in the area cannot be assured.

GRAPHIC: Illustrations: photos during history of Indian Point

Source: All Sources > News > By Individual Publication > N > **The New York Times**
Terms: **ryan and insane and robert and power and plant** (Edit Search)
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	:
	:
ENTERGY CORPORATION	:
	:
(Indian Point Nuclear Power Station,	:
Units No. 2 and 3;	:
Facility Operating Licenses DPR-26 and	:
DPR-64)	:

DECLARATION OF 7 DECEMBER 2001
BY DR. GORDON THOMPSON IN SUPPORT OF
A PETITION BY RIVERKEEPER, INC.

I, Gordon Thompson, declare as follows:

I. INTRODUCTION

(I-1) I am the executive director of the Institute for Resource and Security Studies (IRSS), a nonprofit, tax-exempt corporation based in Massachusetts. Our office is located at 27 Ellsworth Avenue, Cambridge, MA 02139. IRSS was founded in 1984 to conduct technical and policy analysis and public education, with the objective of promoting peace and international security, efficient use of natural resources, and protection of the environment. A statement of my professional qualifications is provided in Section II, below.

(I-2) I have been retained by Riverkeeper, Inc. (hereafter referred to as "Riverkeeper") as an expert witness and adviser in connection with the potential threat posed to public health and safety by the Indian Point nuclear power station, Units 1, 2 and 3. In that capacity, I have prepared this declaration.

(I-3) The purpose of this declaration is to support a petition to the US Nuclear Regulatory Commission (NRC) by Riverkeeper, requesting a proceeding to modify, suspend or revoke the operating licenses for Indian Point Units 2 and 3. In its petition, Riverkeeper identifies the threat of a terrorist attack on the Indian Point station as a new, site-specific, hazardous condition that was not previously considered in the licensing of Indian Point Units 2 and 3, and was not accounted

for in the design-basis threat that the NRC adopted for these units. In this declaration, I address the threat of a terrorist attack within a wider context. I discuss the vulnerability of the Indian Point station to a variety of influences, including acts of malice or insanity. Terrorist attacks represent one category of potential acts of malice or insanity to which the station may be vulnerable. Also, in discussing the vulnerability of the Indian Point station, I consider the vulnerability of the spent fuel storage facility at Unit 1.

(I-4) In addition to discussing the vulnerability of the Indian Point station to a variety of influences, this declaration discusses options whereby the station's vulnerability could be reduced. Moreover, this declaration sets forth requirements for a thorough, credible review of the vulnerability of the Indian Point station, and of the options whereby the station's vulnerability could be reduced. In addressing vulnerability, I focus on the potential for a large release of radioactive material from the Indian Point station to the environment.

(I-5) This declaration has eleven sections. After this introduction (Section I), the declaration addresses my professional qualifications (Section II). Then, the declaration provides some information (Section III) about the Indian Point nuclear power station. Section IV provides a generic discussion of the potential for occurrence of a reactor accident or the onset of self-sustaining, exothermic oxidation reactions in a spent fuel pool; the latter event is referred to here, for simplicity, as a "pool fire". This is followed by a generic discussion (Section V) of the history of, and potential for, acts of malice or insanity at nuclear facilities. The vulnerability of the Indian Point station to a range of events -- including acts of malice or insanity -- is discussed in Section VI. A brief discussion of the offsite consequences of a reactor accident or pool fire at the Indian Point station is provided in Section VII. Options for reducing the vulnerability of the Indian Point station are discussed in Section VIII. Then follows a discussion (Section IX) about the identification and management of sensitive information relating to the vulnerability of nuclear facilities. Section X sets forth some requirements for a credible review of the vulnerability of the Indian Point station, and of the options whereby the station's vulnerability can be reduced. Conclusions are presented in Section XI.

II. MY PROFESSIONAL QUALIFICATIONS

(II-1) I am an expert in the technical analysis of safety and environmental issues related to nuclear facilities. My Curriculum Vitae is provided here as Attachment A.

(II-2) I received an undergraduate education in science and mechanical engineering at the University of New South Wales, in Australia. Subsequently, I pursued graduate studies at Oxford University and received from that institution a Doctorate of Philosophy in mathematics in 1973, for analyses of plasmas undergoing thermonuclear fusion. During my graduate studies I was associated with the fusion research program of the UK Atomic Energy Authority. My undergraduate and graduate work provided me with a rigorous education in the methodologies and disciplines of science, mathematics, and engineering.

(II-3) Since 1977, a significant part of my work has consisted of technical analyses of safety and environmental issues related to nuclear facilities. These analyses have been sponsored by a variety of nongovernmental organizations and local, state and national governments, predominantly in North America and Western Europe. Drawing upon these analyses, I have provided expert testimony in legal and regulatory proceedings, and have served on committees advising US government agencies. To illustrate my expertise, I provide in the following paragraphs some details of my experience.

(II-4) I have conducted, directed, and/or participated in a number of studies that evaluated aspects of the design and operation of nuclear facilities with respect to severe accident probabilities and consequences. These include generic studies and studies of individual facilities. For instance, with respect to generic studies on the potential for severe accidents at nuclear power plants, I was co-investigator in a study by the Union of Concerned Scientists on the "source term" issue -- the potential for release of radioactive material to the environment.[1] Also, I was one of a team of four scientists who prepared, for Greenpeace International, a comprehensive critique of the state of the art of probabilistic risk assessment (PRA) for nuclear power plants.[2] Our report noted that acts of malice, such as sabotage and acts of war, are not considered in PRAs, despite a history of malicious acts at many nuclear facilities. In addition, I conducted analysis on the relevance of PRA to emergency response planning, as part of a study on emergency planning for nuclear power plant accidents.[3] All of these studies required me to be highly familiar with the design and operation of

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[1] Steven Sholly and Gordon Thompson, The Source Term Debate (Cambridge, Massachusetts: Union of Concerned Scientists, January 1986).
[2] H Hirsch et al, IAEA Safety Targets and Probabilistic Risk Assessment (Hannover, Germany: Gesellschaft fur Okologische Forschung und Beratung mbH, August 1989).
[3] D Golding et al, Preparing for Nuclear Power Plant Accidents (Boulder, Colorado: Westview Press, 1995).

nuclear power plants, as well as the characteristics of probabilistic risk assessment.

(II-5) I have also done considerable work on the risks posed by individual nuclear facilities. In addition to performing the studies described elsewhere in this declaration, I have studied the risks posed by the Seabrook and Three Mile Island plants (USA), the Darlington and Pickering stations (Canada), the Sizewell B station (UK) and the Dukovany plant (Czech Republic). All of these studies required me to become familiar with the relevant details of the design and operation of the facilities involved.

(II-6) To a significant degree, my work has been accepted or adopted by relevant governmental agencies. During the period 1978-1979, for example, I served on an international review group commissioned by the government of Lower Saxony (a state in Germany) to evaluate a proposal for a nuclear fuel cycle center at Gorleben. I led the subgroup that examined accident risks and identified alternative options with lower risk.[4] One of the risk issues that I identified and analysed was the potential for self-sustaining, exothermic oxidation reactions of fuel cladding in a high-density spent fuel pool if water is lost from the pool. Hereafter, for simplicity, this event is referred to as a "pool fire".[5] In examining the potential for a pool fire, I identified partial loss of water as a more severe condition than total loss of water. I identified a variety of events that could cause a loss of water from a pool, including aircraft crash, sabotage, terrorism and acts of war. Also, I identified and described alternative fuel storage options with lower risk; these lower-risk options included design features such as spatial separation, natural cooling and underground vaults. The Lower Saxony government accepted my findings about the risk of a pool fire, and ruled in May 1979 that high-density pool storage of spent fuel was not an acceptable option at Gorleben. As a direct result, policy throughout Germany has been to use dry storage in casks, rather than high-density pool storage, for away-from-reactor storage of spent fuel.

(II-7) My work has also influenced decisionmaking by safety officials in the U.S. Department of Energy (DOE). During the period 1986-1991, I was commissioned

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[4] Jan Beyea, Yves Lenoir, Gene Rochlin and Gordon Thompson (subgroup chair), Report of the Gorleben International Review, Chapter 3: Potential Accidents and their Effects, submitted (in German) to the Government of Lower Saxony, March 1979.

[5] At water-cooled reactors, such as the Indian Point Unit 2 and Unit 3 reactors, the fuel cladding is made from a zirconium alloy that can enter into a vigorous exothermic oxidation reaction with either air or steam. For simplicity, this reaction can be referred to as a "fire".

by environmental groups to assess the safety of the military production reactors at the Savannah River Site, and to identify and assess alternative options for the production of tritium for the US nuclear arsenal. Initially, much of the relevant information was classified or otherwise inaccessible to the public. Nevertheless, I addressed safety issues through analyses that were recognized as accurate by nuclear safety officials at DOE. I eventually concluded that the Savannah River reactors could not meet the safety objectives set for them by DOE.[6] DOE subsequently reached the same conclusion, and scrapped the reactors. The current national policy for tritium production is to employ commercial reactors, an option that I had concluded was technically attractive but problematic from the perspective of nuclear weapons proliferation.

(II-8) In 1977, and again during the period 1996-2000, I examined the safety of nuclear fuel reprocessing and liquid high-level radioactive waste management facilities at the Sellafield site in the UK. My investigation in the latter period was supported by consortia of local governments in Ireland and the UK, and I presented my interim findings at briefings in the UK and Irish parliaments in 1998. I identified safety issues that were not addressed in any publicly available literature about the Sellafield site.[7] As a direct result of my investigation, the UK Nuclear Installations Inspectorate (NII) required the operator of the Sellafield site -- British Nuclear Fuels (BNFL) -- to conduct extensive safety analyses. These analyses confirmed the significance of the safety issues that I had identified, and in January 2001 the NII established a legally binding schedule for reduction of the inventory of liquid high-level radioactive waste at Sellafield.[8] The NII took this action in recognition of the grave offsite consequences of a release to the environment from the tanks in which liquid high-level waste is stored. I had identified a variety of events that could cause such a release, including acts of malice or insanity.

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[6] Gordon Thompson and Steven C Sholly, No Restart for K Reactor (Cambridge, Massachusetts: Institute for Resource and Security Studies, October 1991).

[7] Gordon Thompson, High Level Radioactive Liquid Waste at Sellafield: Risks, Alternative Options and Lessons for Policy (Cambridge, Massachusetts: Institute for Resource and Security Studies, June 1998).

[8] Nuclear Installations Inspectorate, "Specification Issued under Licence Condition 32(4) for the Limitation of the Accumulation or Storage of Liquid High Level Radioactive Waste in B215. Licence Instrument 343. January 2001."

(II-9) In May 2000 I completed a study for Greenpeace International on the hazard potential of the La Hague site in France.[9] Nuclear fuel reprocessing and related activities are conducted at this site. The operator of the site -- COGEMA -- is authorised to store 14,000 tonnes of spent fuel in high-density pools at La Hague, and proposes to increase the capacity of these pools to 17,600 tonnes. My study described the potential for a pool fire at La Hague, and identified events -- including acts of malice or insanity -- that could lead to a pool fire. One of the findings of my study was that neither COGEMA nor the French government had a thorough understanding of La Hague's hazard potential, including the potential for a pool fire. Subsequent to the terrorist events of 11 September 2001 in New York and Washington, media exposure brought La Hague's hazard potential to the attention of the French government. During October 2001 the French government deployed anti-aircraft missiles at La Hague.

(II-10) As stated in paragraph II-6, I determined in the period 1978-1979 that partial loss of water from a high-density spent fuel pool is a more severe condition than total loss of water. This is because convective heat transfer is suppressed by the presence of residual water at the base of the fuel assemblies. During any scenario for loss of water from a spent fuel pool, there will be a period of time during which residual water is present. As a result, comparatively old fuel -- potentially including fuel aged 10 or more years after discharge from a reactor -- can ignite if water is lost from a high-density spent fuel pool. The NRC Staff failed, for more than two decades, to understand this point. An illustration of the Staff's lack of understanding was provided by its statements during a license amendment proceeding in regard to the expansion of spent fuel pool capacity at the Harris nuclear power plant. I served as an expert witness for Orange County, North Carolina, the intervenor in this proceeding. In filings during March and April 2000, the Staff repeatedly disparaged my statements that comparatively old fuel can ignite. A few months later, however, the Staff adopted my position. In a report dated October 2000, but not published until January 2001, the Staff recognized that the flow of air to exposed fuel assemblies could be blocked by the presence of collapsed structures -- which might be attributable, for example, to a cask drop or an earthquake -- or by the

[9] Gordon Thompson, Hazard Potential of the La Hague Site: An Initial Review (Cambridge, Massachusetts: Institute for Resource and Security Studies, May 2000).

presence of residual water.[10] The Staff analyzed the heat transfer implications of flow blockage and concluded:[11]

> "While the February 2000 [draft] study indicated that for the cases analyzed a required decay time of 5 years would preclude a zirconium fire, the revised analyses show that it is not feasible, without numerous constraints, to define a generic decay heat level (and therefore decay time) beyond which a zirconium fire is not physically possible."

(II-11) On numerous occasions, I have drawn attention in my writings and oral presentations to the vulnerability of nuclear facilities to acts of malice or insanity. I have pointed out that PRAs do not address acts of malice or insanity, with the result that a PRA can, at best, provide a lower bound to the probability of a release of radioactive material.[12] In 1996 I wrote a generic report on war and terrorism as risk factors for nuclear power plants.[13] Among other findings, this report noted that an act of war or terrorism at a nuclear power plant might have as its primary target the spent fuel stored at the plant, rather than the reactor. The report concluded with a statement that supports Riverkeeper's concern about the threat of a terrorist attack at Indian Point. My statement was:

> "Public debate about the future operation of existing nuclear power plants, and the construction of new plants, should be broadened to encompass the possible involvement of nuclear plants in war or terrorism."

III. THE INDIAN POINT NUCLEAR POWER STATION

(III-1) The Indian Point nuclear power station has three units, all owned by Entergy. Unit 1 had a rated power of 590 MW (thermal) and operated from 1962 to 1974.[14] Unit 2 has a rated power of 2,760 MW (thermal), commenced operating in 1974, and remains operational. Unit 3 has a rated power of 2,760 MW (thermal), commenced operating in 1976, and remains operational. Unit 2 and Unit 3 each employ a four-loop Westinghouse pressurised-water reactor

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[10] Timothy Collins et al (authors are all from the NRC Staff), Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, October 2000.
[11] Ibid, page 2-1.
[12] The strengths and weaknesses of PRA methodology are discussed in Hirsch et al, August 1989 (op cit).
[13] Gordon Thompson, War, Terrorism and Nuclear Power Plants (Canberra: Peace Research Centre, Australian National University, October 1996).
[14] T J Thompson and J G Beckerley (editors), The Technology of Nuclear Reactor Safety, Volume 2 (Cambridge, Massachusetts: MIT Press, 1973), Table 4-1.

(PWR) with a large, dry containment. The reactor cores of Unit 2 and Unit 3 each contain 193 fuel assemblies.[15]

(III-2) Unit 2 and Unit 3 are each equipped with one spent fuel pool. The capacity of the Unit 2 pool is 1,374 fuel assemblies, while the capacity of the Unit 3 pool is 1,345 fuel assemblies.[16] It can be assumed that both pools employ high-density racks. As of November 1998, the Unit 2 pool contained 917 fuel assemblies, while the Unit 3 pool contained 672 fuel assemblies.[17] It is likely that the pool inventories have increased since November 1998.

(III-3) The inventory of cesium-137 in a nuclear facility is a useful indicator of the potential, long-term consequences of a release of radioactive material from that facility. Cesium-137 is a radioactive isotope with a half-life of 30 years. This isotope accounts for most of the offsite radiation exposure that is attributable to the 1986 Chernobyl reactor accident, and for about half of the radiation exposure that is attributable to fallout from nuclear weapons tests in the atmosphere.[18] Cesium is a volatile element that would be liberally released during a pool fire. An NRC study has concluded that a generic estimate of the release fraction of cesium isotopes during a pool fire -- that is, the fraction of the pool's inventory of cesium isotopes that would reach the atmosphere -- is 100 percent.[19] It is reasonable to assume such a high release fraction because cesium is volatile, because a fire in a high-density pool, once initiated, would eventually involve all of the fuel in the pool, and because pool buildings are not designed as containment structures.

(III-4) The inventory of cesium-137 in the Indian Point pools can be readily estimated. Three parameters govern this estimate -- the number of spent fuel assemblies, their respective burnups, and their respective ages after discharge. I have conducted such an estimate, assuming a representative, uniform burnup of 46 GW-days per tonne. The results are provided in the following paragraph.

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[15] Jay R Larson, Systems Analysis Handbook, NUREG/CR-4041 (Washington, DC: NRC, November 1985), Table A-2.
[16] US Nuclear Regulatory Commission, "Reactor Spent Fuel Storage", table downloaded from NRC Web site, 30 May 2001.
[17] Ibid.
[18] US Department of Energy, Health and Environmental Consequences of the Chernobyl Nuclear Power Plant Accident, DOE/ER-0332 (Washington, DC: DOE. June 1987).
[19] V L Sailor et al, Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82, NUREG/CR-4982 (Washington, DC: NRC, July 1987).

(III-5) The 917 fuel assemblies that were in the Unit 2 pool in November 1998 now contain about 42 million Curies (460 kilograms) of cesium-137. The 672 fuel assemblies that were in the Unit 3 pool in November 1998 now contain about 31 million Curies (350 kilograms) of cesium-137. Additional amounts of cesium-137 would be present in any fuel assemblies that have been added to these pools since November 1998. The cores of the Indian Point Unit 2 and Unit 3 reactors each contain about 6 million Curies (67 kilograms) of cesium-137.

(III-6) For comparison with the inventory estimates in paragraph III-5, note that the Chernobyl reactor accident of 1986 released about 2.4 million Curies (27 kilograms) of cesium-137 to the atmosphere. That release represented 40 percent of the Chernobyl reactor core's inventory of 6 million Curies (67 kg) of cesium-137.[20] Also, atmospheric testing of nuclear weapons led to the deposition of about 20 million Curies (220 kilograms) of cesium-137 across the land and water surfaces of the Northern Hemisphere.[21]

IV. REACTOR ACCIDENTS AND POOL FIRES: A GENERIC DISCUSSION

(IV-1) Nuclear reactors around the world have experienced a number of accidents, including the 1979 Three Mile Island accident and the 1986 Chernobyl accident. For reactors such as the PWRs at Indian Point Units 2 and 3, there has been a three-decade analytic effort, drawing from operating experience and supported to some extent by experiments, to estimate the probabilities and characteristics of potential future accidents. A probabilistic risk assessment (PRA) for a particular reactor can use the analytic techniques and knowledge that have been developed through this effort, in order to estimate the probabilities and offsite consequences of potential accident scenarios at that reactor. The state of the PRA art is exemplified by NUREG-1150, a study that was performed by the NRC.[22]

(IV-2) PRAs, such as NUREG-1150, consider a variety of types of accident-initiating event. These types of event include equipment failure, operator error and natural hazards (e.g., an earthquake). Explosions and aircraft impacts have been considered in some PRAs. However, PRAs have not considered acts of

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[20] Allan S Krass, Consequences of the Chernobyl Accident (Cambridge, Massachusetts: Institute for Resource and Security Studies, December 1991).

[21] US Department of Energy, June 1987 (op cit).

[22] US Nuclear Regulatory Commission, Severe Accident Risks: An Assessment for Five US Nuclear Power Plants, NUREG-1150 (Washington, DC: NRC, December 1990).

malice or insanity, such as the deliberate crash of a commercial aircraft on a nuclear power station. This omission means that PRAs have underestimated the probability of reactor accidents. It also means that PRAs have failed to identify accident scenarios that could arise from acts of malice or insanity, but not from other causative events. Note that I am using the word "accident" to encompass any scenario, including a scenario initiated by an act of malice or insanity, that leads to a substantial release of radioactive material to the environment.

(IV-3) In the 1970s, the spent fuel pools of US nuclear power stations were typically equipped with low- or medium-density, open-frame racks. If water were partially or totally lost from such a pool, air or steam could circulate freely throughout the racks, providing cooling to the spent fuel. By contrast, high-density racks -- such as those I assume to be now employed in the Indian Point Unit 2 and Unit 3 pools -- have a closed structure. To suppress criticality, each fuel assembly is surrounded by solid, neutron-absorbing panels, and there is little or no gap between the panels of adjacent cells. This configuration allows only one mode of circulation of air and steam around a fuel assembly -- vertically upward within the confines of the neutron-absorbing panels.

(IV-4) If water is totally lost from a high-density pool, air will pass downward through available gaps such as the gap between the pool wall and the outer faces of the racks, will travel horizontally across the base of the pool, will enter each rack cell through a hole in its base, and will rise upward within the cell, providing cooling to the spent fuel assembly in that cell. If the fuel has been discharged from the reactor comparatively recently, the flow of air may be insufficient to remove all of the fuel's decay heat. In that case, the temperature of the fuel cladding may rise to the point where a self-sustaining, exothermic oxidation reaction with air will begin. In simple terms, the fuel cladding -- which is made of zirconium alloy -- will begin to burn. The zirconium alloy cladding can also enter into a self-sustaining, exothermic oxidation reaction with steam. Other exothermic oxidation reactions can also occur in a pool if water is lost. For simplicity, the occurrence of one or more of the possible reactions is referred to here as a "pool fire".

(IV-5) In many scenarios for loss of water from a pool, the flow of air that is described in paragraph IV-4 will be blocked. For example, an earthquake or the drop of a shipping cask may distort rack structures, thereby blocking air flow. Alternatively, an earthquake, aircraft impact or explosion may cause objects -- for example, the roof of the fuel handling building -- to fall into the pool, leading to a blockage of air flow. The presence of residual water in the bottom of the pool would also block air flow. In most scenarios for loss of water, residual water will

be present for significant periods of time. Blockage of air flow, for whatever reason, will lead to ignition of fuel that has been discharged from a reactor for long periods -- potentially 10 years or longer.[23] The NRC Staff failed to understand this point for more than two decades (see paragraph II-10).

(IV-6) The NRC Staff has prepared or sponsored a number of generic, technical studies related to the potential for a pool fire. The first of these studies was conducted for the Staff by Sandia Laboratories in 1979.[24] The most recent was a Staff study, dated October 2000 but published in January 2001, that addressed the risk of a pool fire at a nuclear power station undergoing decommissioning.[25] In a February 1999 report for Orange County, North Carolina, I reviewed some of the Staff analyses conducted prior to February 1999.[26] I reviewed the Staff's October 2000 study in comments submitted to the NRC Commissioners in February 2001.[27]

(IV-7) Technical documents related to the potential for a pool fire were generated in the course of a license amendment proceeding (see paragraph II-10) in regard to the expansion of spent fuel pool capacity at the Harris nuclear power station. I prepared a report for Orange County, the intervenor in this proceeding.[28] The NRC Staff's principal technical document was an affidavit by members of the Staff.[29] The principal technical document proffered by the

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[23] The role of residual water in promoting ignition of old fuel is discussed in: Gordon Thompson, Risks and Alternative Options Associated with Spent Fuel Storage at the Shearon Harris Nuclear Power Plant (Cambridge, Massachusetts: Institute for Resource and Security Studies, February 1999), Appendix D.

[24] Allan S Benjamin et al, Spent Fuel Heatup Following Loss of Water During Storage, NUREG/CR-0649 (Washington, DC: NRC, March 1979).
[25] Collins et al, October 2000 (op cit).
[26] Thompson. February 1999 (op cit), Appendix D.
[27] Gordon Thompson, Comments on the NRC Staff's Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (Cambridge, Massachusetts: Institute for Resource and Security Studies, 19 February 2001).

[28] Gordon Thompson, The Potential for a Large, Atmospheric Release of Radioactive Material from Spent Fuel Pools at the Harris Nuclear Power Plant: The Case of a Pool Release Initiated by a Severe Reactor Accident (Cambridge, Massachusetts: Institute for Resource and Security Studies, 20 November 2000).

[29] ASLBP No. 99-762-02-LA, "Affidavit of Gareth W Parry, Stephen F LaVie, Robert L Palla and Christopher Gratton in Support of NRC Staff Brief and Summary of Relevant Facts, Data and Arguments upon which the Staff Proposes to Rely at Oral Argument on Environmental Contention EC-6", 20 November 2000.

licensee -- Carolina Power and Light (CP&L) -- was a document prepared by ERIN Engineering.[30] Each of these documents was limited in scope, in the sense that the Atomic Safety and Licensing Board (ASLB) had ordered the three parties to confine their analyses to a single scenario for a pool fire.[31] In the postulated scenario, a severe accident at the Harris reactor would contaminate the Harris site with radioactive material to an extent that would preclude actions needed to supply cooling and makeup to the Harris pools. The Harris station has one reactor and four pools. Two pools were in use at high density prior to the proceeding, and the proceeding addressed the activation of the two remaining pools, also at high density.

(IV-8) In its October 2000 report (see paragraph II-10), the NRC Staff conceded that comparatively long-discharged fuel can ignite in the event of water loss from a high-density pool. In the Harris proceeding, the Staff made the same concession. Staff members stated that loss of water from pools containing fuel aged less than 5 years "would almost certainly result in an exothermic reaction", and also stated: "Precisely how old the fuel has to be to prevent a fire is still not resolved."[32] The Staff assumed -- conservatively, in its view -- that a fire would be inevitable if the water level fell to the top of the racks.[33] Thus, the Staff has conceded that the potential for a pool fire is equivalent to the potential for a loss of water down to the top of the racks.

(IV-9) Partial or total loss of water from a spent fuel pool could occur through leakage, evaporation, siphoning, pumping, displacement by objects falling into the pool, or overturning of the pool. These modes of loss of water could arise from events, alone or in combination, that include: (a) acts of malice or insanity by persons within or outside the station boundary; (b) an aircraft impact, with or without an accompanying fuel-air explosion or fire; (c) an earthquake; (d) dropping of a fuel transfer cask or shipping cask; (e) a severe accident at a nearby reactor or spent fuel pool which, through the spread of radioactive material and other influences, precludes the ongoing provision of cooling and/or water makeup to the affected pool; and (f) an explosion inside or outside the station buildings.

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[30] ERIN Engineering and Research Inc, "Technical Input for Use in the Matter of Shearon Harris Spent Fuel Pool Before the Atomic Safety and Licensing Board (Docket No. 50-400-LA)", November 2000.
[31] ASLBP No. 99-762-02-LA, "Memorandum and Order (Ruling on Late-Filed Environmental Contention)", 7 August 2000.
[32] Parry et al, November 2000 (op cit), paragraph 29.
[33] Ibid, paragraphs 29 and 124.

(IV-10) Neither the NRC nor any other entity has performed a study (of the potential for a pool fire) that addresses all of the modes of water loss and causative events that are mentioned in paragraph IV-9. Such a study could be performed by extending the analytic techniques that are currently available in the field of PRA. A credible study would consider all of the modes of loss of water from a pool that are mentioned in paragraph IV-9, all of the events that could cause a loss of water that are mentioned in paragraph IV-9, and all physically realisable combinations of causative events and modes of water loss. The study would not be credible if it arbitrarily considered only a subset of the physically realisable combinations of causative events and modes of water loss.[34] Moreover, a credible study would differ from current PRA practice in that it would consider causative events -- including acts of malice and insanity -- for which the estimation of probability has been regarded as difficult or impossible. If a credible, numerical estimate of the probability of a causative event cannot be made, the foreseeability of that event should be addressed through qualitative analysis.

(IV-11) Various studies prepared by or sponsored by the NRC Staff have addressed selected scenarios for a loss of water from a spent fuel pool. For example, a Staff study -- NUREG-1353 -- has drawn upon other literature to provide a generic estimate that the probability of a loss of water from the dropping of a shipping cask is 3.1 per 100 million reactor-years.[35] This estimate assumes that the conditional probability of a loss of water, given the dropping of a cask, is 0.1, with an uncertainty range of 0.01 to 1.0. Acts of malice or insanity are not considered. An alternative formulation of this estimate would be that the probability (per reactor-year) of water loss from a cask drop = [3.1 per 1,000 million to 3.1 per 10 million] + [(0.01 to 1.0) x (the probability that a cask will drop due to an act of malice or insanity)]. In simpler terms, arranging for the drop of a cask could be an option that appeals to a malicious or insane person.

(IV-12) NUREG-1353 also provides an estimate that the probability of a loss of water from a pool due to aircraft impact is 6.0 per 1,000 million reactor-years as a best estimate, with an upper bound of 2.0 per 100 million reactor-years.[36] The same numbers are provided elsewhere in NUREG-1353 as the "hit frequency" of

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[34] The ASLB in the Harris proceeding ordered the parties to analyse only one scenario for water loss from the Harris pools (see paragraph IV-7). Such an arbitrary limitation of the scope of a study guarantees that its findings will provide, at best, a lower bound to the potential for a pool fire.

[35] E D Throm, Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools", NUREG-1353 (Washington, DC: NRC, April 1989), page 4-14.

[36] Ibid, Table 4.7.1 (at page 4-36).

aircraft impact.[37] Thus, NUREG-1353 assumes that the conditional probability of a loss of water from a spent fuel pool, given an aircraft impact, is 1.0 (100 percent). This assumption may have been made thoughtlessly, because NRC analyses typically give little attention to threats that are judged to have very low probability.

(IV-13) The NRC Staff's October 2000 report includes a crude, generic analysis of the conditional probability that aircraft impact will cause a loss of water from a spent fuel pool.[38] The pool is assumed to have a 5-ft-thick reinforced concrete wall. Impacting aircraft are divided into the categories "large" (weight more than 5.4 tonnes) and "small" (weight less than 5.4 tonnes). The Staff estimates that the conditional probability of penetration of the pool wall by a large aircraft is 0.45, and that 50 percent of penetration incidents involve a loss of water which exposes fuel to air. Thus, the Staff estimates that, for impact of a large aircraft, the conditional probability of a loss of water sufficient to initiate a pool fire is 0.23 (23 percent).

(IV-14) The abovementioned Harris proceeding considered a pool fire scenario (see paragraph IV-7) in which the release of radioactive material during a reactor accident precludes actions that are needed to provide cooling and makeup to spent fuel pools. My analysis found that the minimum value for the best estimate of a pool fire, for this scenario, is 1.6 per 100 thousand reactor-years. This estimate did not account for acts of malice, degraded standards of plant operation, or gross errors in design, construction or operation.[39] The NRC Staff estimated, for the same scenario, that the probability of a pool fire is on the order of 2 per 10 million reactor-years.[40] The ASLB accepted the Staff's estimate, thereby concluding that the postulated scenario is "remote and speculative", and terminated the proceeding.[41] In another declaration, I have described numerous deficiencies in the ASLB's ruling.[42]

(IV-15) The Harris fuel handling building contains four pools. In the Harris proceeding, the NRC Staff stated its view that the onset of a pool fire in two of the pools would preclude the provision of cooling and makeup to the other two

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[37] Ibid, page 4-14.
[38] Collins et al, October 2000 (op cit), page 3-23 and Appendix 2D.
[39] Thompson, November 2000 (op cit), page 43.
[40] Parry et al, November 2000 (op cit), paragraph 251.
[41] ASLBP No. 99-762-02-LA, "Memorandum and Order (Denying Request for Evidentiary Hearing and Terminating Proceeding)", 1 March 2001.
[42] US Court of Appeals for the District of Columbia Circuit, No. 01-1246, "Declaration of 31 May 2001 by Dr Gordon Thompson in Support of Orange County's Stay Motion".

pools.[43] This view was not supported by any analysis or rationale. ERIN Engineering, on behalf of CP&L, expressed the opinion: "The consequences of loss of water inventory in pools A and B could in turn adversely impact both access and further prevention actions related to pools C and D."[44] Again, this opinion was not supported by any analysis or rationale. It seems clear, however, that the Staff and ERIN Engineering are in agreement with one of my findings, which was based on analysis. My finding was that the onset of a fire in one or more pools would, through the creation of radioactive contamination of the site and other influences, preclude the provision of cooling and makeup to nearby pools, thereby leading to the onset of fires in the nearby pools.[45]

(IV-16) The observations set forth in the preceding paragraphs of Section IV lead to at least five broad, generic conclusions about reactor accidents and pool fires. First, the probabilities and characteristics of potential reactor accidents have been estimated through PRAs. Second, PRAs have not considered acts of malice or insanity, and therefore provide an incomplete picture of accident risk. Third, a variety of events, including acts of malice or insanity, could lead to a loss of water from a spent fuel pool, thereby initiating a pool fire. Fourth, the potential for pool fires has not been subjected to the depth of analysis that has been devoted to the potential for reactor accidents. Fifth, NRC documents indicate that a pool fire could be readily initiated by a range of events including the drop of a fuel cask, an aircraft impact, or a fire at a nearby pool.

V. ACTS OF MALICE OR INSANITY AT NUCLEAR FACILITIES: A GENERIC DISCUSSION

(V-1) For two decades or more it has been clear to many people that nuclear power stations and other nuclear facilities are potential targets of acts of malice or insanity, including highly destructive acts. The NRC has repeatedly rebuffed suggestions by members of the public that this threat be given the depth of analysis that would be expected, for example, in an environmental impact statement (EIS). This history is illustrated by a September 1982 ruling by the ASLB in the operating license proceeding for the Harris station. The intervenor, Wells Eddleman, had proffered a contention alleging, in part, that the station's safety analysis was deficient because it did not consider the "consequences of

[43] Parry et al, November 2000 (op cit), paragraph 29.
[44] ERIN Engineering, November 2000 (op cit), page 2-36.
[45] Thompson, November 2000 (op cit), page 40.

terrorists commandeering a very large airplane.....and diving it into the containment." In rejecting this contention the ASLB stated:[46]

> "This part of the contention is barred by 10 CFR 50.13. This rule must be read *in pari materia* with 10 CFR 73.1(a)(1), which describes the "design basis threat" against which commercial power reactors *are* required to be protected. Under that provision, a plant's security plan must be designed to cope with a violent external assault by "several persons," equipped with light, portable weapons, such as hand-held automatic weapons, explosives, incapacitating agents, and the like. Read in the light of section 73.1, the principal thrust of section 50.13 is that military style attacks with heavier weapons are not a part of the design basis threat for commercial reactors. Reactors could not be effectively protected against such attacks without turning them into virtually impregnable fortresses at much higher cost. Thus Applicants are not required to design against such things as artillery bombardments, missiles with nuclear warheads, or kamikaze dives by large airplanes, despite the fact that such attacks would damage and may well destroy a commercial reactor."

(V-2) In the statement quoted in paragraph V-1, the ASLB correctly described the design basis for US nuclear power stations. However, other design bases are possible. In the early 1980s the reactor vendor ASEA-Atom developed a preliminary design for an "intrinsically safe" commercial reactor known as the PIUS reactor. The design basis for the PIUS reactor included events such as equipment failures, operator errors and earthquakes, but also included: (a) takeover of the plant for one operating shift by knowledgeable saboteurs equipped with large amounts of explosives; (b) aerial bombardment with 1,000-pound bombs; and (c) abandonment of the plant by the operators for one week.[47] It seems likely that this design basis would also provide protection against the impact of a large, fuel-laden aircraft. Clearly, ASEA-Atom foresaw a world in which acts of malice could pose a significant threat to nuclear power stations.

(V-3) There is a rich history of events which shows that acts of malice pose a significant threat to nuclear power stations around the world. Many of these events, up to 1996, are summarised in a report that I prepared.[48] Consider some

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[46] Carolina Power and Light Co. (Shearon Harris Nuclear Power Plant, Units 1 and 2), LBP-82-119A, 16 NRC 2069, 2098 (1982), (emphasis in original).
[47] K Hannerz, Towards Intrinsically Safe Light Water Reactors (Oak Ridge, Tennessee: Institute for Energy Analysis, February 1983).
[48] Thompson, October 1996 (op cit).

examples. Nuclear stations under construction in Iran were repeatedly bombed from the air by Iraq in the period 1984-1987. Yugoslav Air Force fighters made a threatening overpass of the Krsko nuclear station in Slovenia -- which was operating at the time -- a few days after Slovenia declared independence in 1991. So-called research reactors in Iraq were destroyed by aerial bombing by Israel in 1981 and by the United States in 1991. In 1987, Iranian radio threatened an attack by unspecified means on US nuclear stations if the United States attacked launch sites for Iran's Silkworm antiship missiles. Bombs damaged reactors under construction in Spain in 1977 and in South Africa in 1982. Antitank missiles struck and penetrated the containment of a nuclear station under construction in France in 1982. North Korean commandos were killed while attempting to come ashore near a South Korean station in 1985. These and other events illustrate the "external" threat to nuclear power stations. Numerous crimes and acts of sabotage by plant personnel illustrate the "internal" threat.

(V-4) The threat posed to nuclear stations by truck bombs became clearly apparent from an October 1983 attack on a US Marine barracks in Beirut. In a suicide mission, a truck was driven at high speed past a guard post and into the barracks. A gas-boosted bomb on the truck was detonated with a yield equivalent to about 5 tonnes of TNT, destroying the building and killing 241 Marines. In April 1984 a study by Sandia National Laboratories titled "Analysis of Truck Bomb Threats at Nuclear Facilities" was presented to the NRC. According to an NRC summary:[49] "The results show that unacceptable damage to vital reactor systems could occur from a relatively small charge at close distances and also from larger but still reasonable size charges at large setback distances (greater than the protected area for most plants)." Eventually, in 1994, the NRC introduced regulations that require licensees to install defenses (gates, barriers, etc.) against vehicle bombs. The NRC was spurred into taking this action by two incidents in February 1993. In one incident, a vehicle bomb was detonated in a parking garage under the World Trade Center in New York. In the other incident, a man recently released from a mental hospital crashed his station wagon through the security gate of the Three Mile Island nuclear stations and rammed the vehicle under a partly-opened door in the turbine building.

(V-5) The threat of suicidal aircraft attack on symbolic or high-value targets became clearly apparent from three incidents in 1994.[50] In April 1994 a Federal

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[49] T A Rehm, memo to the NRC Commissioners, "Weekly Information Report -- Week Ending April 20, 1984".
[50] Matthew L Wald, "US Failed to Learn From Earlier Hijackings", International Herald Tribune, 4 October 2001, page 6.

Express flight engineer who was facing a disciplinary hearing was travelling as a passenger on a company DC-10. He stormed the cockpit, severely wounded all three members of the crew with a hammer, and tried to gain control of the aircraft. The crew regained control with great difficulty. Federal Express employees said that the flight engineer was planning to crash into a company building. In September 1994 a lone pilot crashed a stolen single-engine Cessna into the grounds of the White House, just short of the President's living quarters. In December 1994 four Algerians hijacked an Air France Airbus 300, carrying 20 sticks of dynamite. The aircraft landed in Marseille, where the hijackers demanded that it be given a large fuel load -- three times more than necessary for the journey -- before flying to Paris. Troops killed the hijackers before this plan could be implemented. French authorities determined that the hijackers planned to explode the aircraft over Paris or crash it into the Eiffel Tower.

(V-6) The incident described in paragraph V-5 involving the Federal Express flight engineer illustrates the vulnerability of industrial systems, including nuclear stations, to "internal" threats. That vulnerability is further illustrated by a number of incidents. In December 2000, Michael McDermott killed seven coworkers in a shooting rampage at an office building in Massachusetts. He had worked at the Maine Yankee nuclear station from 1982 to 1988 as an auxiliary operator and operator before being terminated for exhibiting unstable behavior.[51] In 1997, Carl Drega of New Hampshire stockpiled weapons and killed four people -- including two state troopers and a judge -- on a suicide mission. He had passed security clearances at three nuclear power stations in the 1990s.[52] In October 2000 a former US Army sergeant pleaded guilty to assisting Osama bin Laden in planning the bombing of the US embassy in Nairobi, which occurred in 1998.[53] In June 1999, a security guard at the Bradwell nuclear station in Britain hacked into the plant's computer system and wiped out records. It emerged that he had never been vetted and had two undisclosed criminal convictions.[54] These and other incidents demonstrate clearly that it is foolish to ignore or downplay the "internal" threat of acts of malice or insanity at nuclear stations.

(V-7) The events mentioned in the preceding paragraphs occurred against a background of numerous acts of terrorism around the world. Many of these acts

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[51] Anne Barnard and Ross Kerber, "Web posting tells of suspect's firing from Maine plant", The Boston Globe, 5 January 2001, page A12.
[52] Ibid.
[53] John J Goldman, "Former sergeant admits role in bombings of US embassies", The Boston Globe, 21 October 2000, page A2.
[54] Kevin Maguire, "Security checks tightened after high-level alert", The Guardian, 9 January 2001.

have been highly destructive. US facilities have been targets on many occasions, as illustrated by the bombing of the US embassy in Beirut in 1983, the embassies in Nairobi and Dar es Salaam in 1998, and the USS Cole in 2000. There have been repeated warnings that the threat of terrorism is growing and could involve the US homeland. For example, three authors with high-level government experience have written:[55]

> Long part of the Hollywood and Tom Clancy repertory of nightmarish scenarios, catastrophic terrorism has moved from far-fetched horror to a contingency that could happen next month. Although the United States still takes conventional terrorism seriously, as demonstrated by the response to the attacks on its embassies in Kenya and Tanzania in August, it is not yet prepared for the new threat of catastrophic terrorism.

(V-8) A few years ago the US Department of Defense established an advisory commission on national security in the 21st century. This commission -- often known as the Hart-Rudman commission because it was co-chaired by former Senators Gary Hart and Warren Rudman -- issued reports in September 1999, April 2000 and March 2001. The findings in the September 1999 report included the following:[56]

> "America will become increasingly vulnerable to hostile attack on our homeland, and our military superiority will not entirely protect us.....States, terrorists and other disaffected groups will acquire weapons of mass destruction and mass disruption, and some will use them. Americans will likely die on American soil, possibly in large numbers."

(V-9) From the preceding paragraphs in Section V it is clear that the potential for acts of malice or insanity at nuclear power stations -- including highly destructive acts -- has been foreseeable for many years, and has been foreseen. However, the terrorist attacks on the World Trade Center and the Pentagon on 11 September 2001 provided significant new information. These attacks conclusively demonstrated that the threat of highly-destructive acts of malice or insanity is a clear and present danger, and that no reasonable person can regard this threat as remote or speculative. According to recent press reports, US

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[55] A Carter, J Deutch and P Zelikow, "Catastrophic Terrorism", Foreign Affairs, November/December 1998, page 80.
[56] US Commission on National Security/21st Century, New World Coming: American Security in the 21st Century, Phase I report, 15 September 1999, page 4.

authorities possess information suggesting that the hijackers of United Airlines flight 93, which crashed in Pennsylvania on 11 September 2001, were planning to hit a nuclear station.[57] This may be true or false, or the truth may never be known. Whatever the truth is, it would be foolish to regard nuclear stations as immune from attack.

(V-10) The NRC Staff has conceded that it cannot provide a quantitative assessment of the probability of an act of malice at a nuclear power station. In a SECY paper for the NRC Commissioners, the Staff has stated:[58]

> "The staff, as a result of its ongoing work with the Federal national security agencies, has determined that the ability to quantify the likelihood of sabotage events at nuclear power plants is not currently supported by the state-of-the-art in PRA methods and data. The staff also believes that both the NRC and the other government stakeholders would need to conduct additional research and expend significant time and resources before it could even attempt to quantify the likelihood of sabotage events. In addition, the national security agencies, Intelligence Community, and Law Enforcement Agencies do not currently quantitatively assess the likelihood of terrorist, criminal, or other malevolent acts."

(V-11) Although the probability of a terrorist attack cannot be assessed quantitatively, it can be assessed qualitatively. From a qualitative perspective, the probability of a terrorist attack within the US homeland appears to be significantly greater in the current period than it was, for example, in the 1980s. There is now a focussed, well-organized and well-financed threat. The United States is taking military action that may provoke further attacks. This new threat environment may persist for many years.

VI. VULNERABILITY OF THE INDIAN POINT STATION

(VI-1) This Section of my declaration addresses the vulnerability of the Indian Point nuclear power station. Units 2 and 3 of the station have operational reactors with attached spent fuel pools. Unit 1 has a spent fuel storage facility.

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[57] Nicholas Rufford, David Leppard and Paul Eddy, "Nuclear Mystery: Crashed plane's target may have been reactor", The Sunday Times, London, 20 October 2001.
[58] William D Travers, memo to the NRC Commissioners, "Policy Issues Related to Safeguards, Insurance, and Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools (WITS 200000126), SECY-01-0100", 4 June 2001, pp 5-6.

In addressing vulnerability, I focus on the potential for a large release of radioactive material to the environment, from one or more of the two operational reactors and three spent fuel storage facilities.

(VI-2) PRA studies have been performed for the Unit 2 and Unit 3 reactors. A review of these studies is beyond the scope of this declaration. However, as noted in paragraph IV-2, these studies have not considered acts of malice or insanity. As a result, they have underestimated the probability of reactor accidents, and have failed to identify accident scenarios that could arise from acts of malice or insanity but not from other causative events.

(VI-3) There has been no PRA-type study to assess the potential for, or consequences of, a spent fuel pool fire at Indian Point. For Units 2 and 3, the potential for onset of a pool fire is equivalent to the potential for loss of water from a pool. As explained in paragraph IV-8, the NRC Staff has conceded that the potential for a fire in a high-density pool is equivalent to the potential for a loss of water down to the top of the racks.

(VI-4) Modes of water loss from a pool, and events that could cause water loss, are set forth in paragraph IV-9, above. Paragraph IV-10 provides specifications for a credible study of the potential for water loss from a pool, pointing out that a credible study would: (a) consider all physically realisable combinations of causative events and modes of water loss; and (b) include acts of malice or insanity in its consideration of causative events. Any order by a licensing tribunal to limit the scope of a study of the potential for water loss, so that only some modes of water loss and some causative events are considered, would render the findings non-credible. Moreover, the ordering of such a limit would suggest that the tribunal is seeking to evade or distort the truth.

(VI-5) A thorough, credible study of the potential for water loss from the Indian Point pools would require substantial effort by a number of investigators. This declaration does not purport to be such a study. Here, I provide an illustrative discussion of some modes of water loss and some causative events. The discussion focusses on acts of malice or insanity, especially aircraft impact. This focus does not imply that other causative events are unimportant.

(VI-6) Paragraph IV-11 points out that an act of malice or insanity could lead to the drop of a shipping cask, causing a loss of water from a pool. In a report about the Harris station, I have sketched a scenario for the deliberate siphoning

of water from a pool.[59] Siphoning could be accomplished by one person equipped with some thick-walled hoses.[60] After the fuel is exposed to air, a fire will begin within a few hours, as explained in paragraph VI-15. A time period sufficient for this scenario could be available if the event were successfully concealed from operators and security staff, or if the pool building were successfully seized and defended by an armed group. Either approach could be accompanied by diversionary activity elsewhere on the station site. Numerous other scenarios could be identified, whereby deliberate actions could potentially lead to a loss of water from an Indian Point pool. Each such scenario deserves detailed analysis, to determine its relevance in the Indian Point context. Detailed information about these scenarios does not necessarily belong in the public domain, as discussed in Section IX of this declaration.

(VI-7) Aircraft impact at the Indian Point site could, through a variety of mechanisms, potentially cause a reactor accident or a loss of water from a spent fuel pool. A scenario involving the hijacking of a commercial aircraft may be less likely now than it was before 11 September 2001, because the airline industry is now aware of this threat. However, according to the physicist Richard Garwin, a scenario involving a rented or stolen cargo aircraft may be no less likely than before 11 September 2001. Garwin, who has served on numerous panels advising the US government, warns that a cargo aircraft may be used against a nuclear plant.[61] Also, one must consider a scenario in which a licensed crew member of a passenger or cargo aircraft engages in a suicide attack. Finally, one must consider the aerial equivalent of a truck bomb, which need not require a large aircraft.

(VI-8) As indicators of the forces that could accompany an aircraft impact, consider the weights and fuel capacities of some typical commercial aircraft.[62] The Boeing 737-300 has a maximum takeoff weight of 56-63 tonnes and a fuel capacity of 20-24 thousand liters. The Boeing 747-400 has a maximum takeoff weight of 363-395 tonnes and a fuel capacity of 204-217 thousand liters. The Boeing 757 has a maximum takeoff weight of 104-116 tonnes and a fuel capacity of 43 thousand liters. The Boeing 767 has a maximum takeoff weight of 136-181 tonnes and a fuel capacity of 63-91 thousand liters.

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[59] Thompson, February 1999 (op cit), Appendix C.

[60] After the water level recedes below the effective siphoning depth, water will be lost due to evaporation. This scenario assumes an absence of pool makeup.

[61] Richard Garwin, "The Many Threats of Terror", The New York Review, 1 November 2001, pp 16-18.

[62] Data here are from Paul Jackson (editor), Jane's All the World's Aircraft, 1996-97 (Alexandria, Virginia: Jane's Information Group, 1996).

(VI-9) Commercial jet fuel typically has a heat of combustion of about 38 MJ per liter. For comparison, 1 kilogram of TNT will yield 4.2 MJ of energy. Thus, complete combustion of 1 liter of jet fuel will yield energy equivalent to that from 9 kilograms of TNT. Complete combustion of 100 thousand liters of jet fuel -- about half the fuel capacity of a Boeing 747-400 -- will yield energy equivalent to that from 900 tonnes of TNT. Thus, the impact of a fuel-laden aircraft can lead to a violent fuel-air explosion. Fuel-air munitions have been developed that yield more than 5 times the energy of their equivalent weight in TNT, and create a blast overpressure exceeding 1,000 pounds per square inch.[63] A fuel-air explosion arising from an aircraft impact will be less efficient than a munition in converting combustion energy into blast, but could nevertheless generate a highly-destructive blast, especially if fuel vapor accumulates in a confined space before igniting.

(VI-10) The NRC Staff report NUREG-1353 assumes (see paragraph IV-12) that the impact of an aircraft on a spent fuel pool will cause a loss of water from the pool with a conditional probability of 100 percent. This assumption is not supported by analysis. The NRC Staff's October 2000 report includes a crude, generic analysis of aircraft impact (see paragraph IV-13), yielding an estimate that, for impact of a large aircraft (weight more than 5.4 tonnes), the conditional probability of a loss of water sufficient to initiate a pool fire is 0.23 (23 percent). This estimate ignores the potential for fuel-air explosions and fires. All of the typical, commercial aircraft mentioned in paragraph VI-8 weigh considerably more than 5.4 tonnes.

(VI-11) A rough indication of the vulnerability of Indian Point Units 2 and 3 to aircraft impact can be obtained from the PRA for the Seabrook station. The Seabrook station and Indian Point Units 2 and 3 all employ 4-loop Westinghouse PWRs with large, dry containments. Thus, PRA findings for Seabrook are roughly indicative of findings for Indian Point Units 2 and 3. The Seabrook PRA finds that any direct impact on the containment by an aircraft weighing more than 37 tonnes will lead to penetration of the containment and a breach in the reactor coolant circuit. Also, the Seabrook PRA finds that a similar impact on the control building or auxiliary building will inevitably lead to a core melt.[64] All of the typical, commercial aircraft mentioned in paragraph VI-8 weigh considerably more than 37 tonnes. Also, the Seabrook PRA does not consider the effects of a

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[63] Tom Gervasi, Arsenal of Democracy (New York: Grove Press, 1977), page 177.
[64] Pickard, Lowe and Garrick Inc, Seabrook Station Probabilistic Safety Assessment, Main Report (Irvine, California: PLG, December 1983), pp 9.3-10 to 9.3-11.

fuel-air explosion and/or fire as an accompaniment to an aircraft impact. Thus, one could plausibly infer from the Seabrook PRA that the impact of a typical, commercial aircraft on Indian Point Unit 2 or Unit 3 could lead to a reactor accident and/or a loss of water from a spent fuel pool, followed by a pool fire.

(VI-12) Analytic techniques are available for estimating the effects that aircraft impact will have on the structures and equipment of a nuclear power station. However, those techniques focus on the kinetic energy of the impacting aircraft. The effects of an accompanying fuel-air explosion and/or fire are given, at best, a crude analysis. A 1982 review by Argonne National Laboratory of the state of the art for aircraft impact analysis stated:[65]

> "Based on the review of past licensing experience, it appears that fire and explosion hazards have been treated with much less care than the direct aircraft impact and the resulting structural response. Therefore, the claim that these fire/explosion effects do not represent a threat to nuclear power plants has not been clearly demonstrated."

My experience in reviewing PRAs and related studies for nuclear facilities leads me to conclude that the Argonne statement remains valid today. Indeed, in view of the large amount of energy that can be liberated in a fuel-air explosion (see paragraph VI-9), I conclude that previous analyses of aircraft impacts may have grossly underestimated the vulnerability of nuclear plants to such impacts.

(VI-13) To my knowledge, there exists no thorough, credible analysis of the vulnerability of any nuclear power station to the impact of a modern commercial aircraft. The conduct of such an analysis would be a necessary part of a review of the vulnerability of the Indian Point station. For each spent fuel pool on the site, the analysis would consider the potential for overturning of the pool, causing water to be spilled, and for the creation of a breach in the pool boundary, causing water to leak out. Also, the analysis would consider the potential for water to be displaced from the pool by blast or the falling of objects into the pool. Finally, the analysis would consider the potential for loss of water by evaporation, which would occur over a period of days if pool cooling and makeup were unavailable.

(VI-14) In the context of the spent fuel pools at the Harris plant, the NRC Staff has conceded (see paragraph IV-15) that a fire in one pool would preclude the

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[65] C A Kot et al, Evaluation of Aircraft Crash Hazards Analyses for Nuclear Power Plants, NUREG/CR-2859 (Washington, DC: NRC, June 1982), page 78.

provision of cooling and makeup to nearby pools. This situation would arise mostly because the initial fire would contaminate the site with radioactive material, generating high radiation fields. An analogous situation could arise in which the release of radioactive material from a damaged reactor precludes the provision of cooling and makeup to nearby pools. For example, an aircraft impact on the Indian Point Unit 2 or Unit 3 reactor could lead to a rapid-onset core melt with an open containment, accompanied by a raging fire. That event would create high radiation fields across the site, potentially precluding any access to the site by personnel. One can envision a variety of "cascading" scenarios, in which there might eventually be fires in the Unit 2 and Unit 3 pools at Indian Point, accompanied by core melt events at Unit 2 and Unit 3. Conceivably, a fire could also occur in the spent fuel storage facility at Unit 1, although the fuel in the Unit 1 reactor used stainless steel cladding, which is less prone to ignite than the zircalloy-clad fuel used in Units 2 and 3.[66] The potential for cascading scenarios should be carefully investigated.

(VI-15) A pool fire could begin comparatively soon after water is lost from a pool. For example, suppose that most of the length of the fuel assemblies is exposed to air, but the flow of air to the base of the racks is precluded by residual water or a collapsed structure. In that event, fuel heatup would be approximately adiabatic. Fuel discharged for 1 month would ignite in less than 2 hours, and fuel discharged for 3 months would ignite in about 3 hours. The fire would then spread to older fuel. Once a fire has begun, it could be impossible to extinguish. Spraying water on the fire would feed an exothermic zirconium-steam reaction which would generate flammable hydrogen. High radiation fields could preclude the approach of firefighters.

(VI-16) Paragraph IV-16, above, summarizes the state of knowledge about the generic potential for reactor accidents and pool fires, and identifies significant deficiencies in knowledge. These generic findings also apply to the Indian Point station, for which there is no thorough, credible analysis of the station's vulnerability. In the absence of such an analysis, one is obliged to resort to judgment. In light of the various studies and factors discussed in this declaration, my judgment is that a pool fire and/or core melt accident at Unit 2 and/or Unit 3 could be caused by a variety of potential acts of malice or insanity, including the deliberate impact of a large aircraft.

VII. OFFSITE CONSEQUENCES OF A REACTOR ACCIDENT OR POOL FIRE AT INDIAN POINT

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[66] Thompson and Beckerley, 1973 (op cit), Table 4-1.

(VII-1) Paragraph III-3 explains that cesium-137 is a useful indicator of the potential, long-term consequences of a release of radioactive material to the environment. The same paragraph shows that it is reasonable to assume that 100 percent of the cesium-137 in a spent fuel pool would be released to the atmosphere in the event of a pool fire. The cesium-137 would be released to the atmosphere in small particles that would travel downwind and be deposited on the ground and other surfaces. The deposited particles would emit intense gamma radiation, leading to external, whole-body radiation doses to exposed persons. Cesium-137 would also contaminate water and foodstuffs, leading to internal radiation doses.

(VII-2) One measure of the scope of radiation exposure attributable to deposition of cesium-137 is the area of land that would become uninhabitable. For illustration, I assume that the threshold of uninhabitability is an external, whole-body dose of 10 rem over 30 years. This level of radiation exposure, which would represent about a three-fold increase above the typical level of background (natural) radiation, was used in the NRC's 1975 Reactor Safety Study as a criterion for relocating populations from rural areas.

(VII-3) A radiation dose of 10 rem over 30 years corresponds to an average dose rate of 0.33 rem per year.[67] The health effects of radiation exposure at this dose level have been estimated by the National Research Council's Committee on the Biological Effects of Ionizing Radiations.[68] This committee has estimated that a continuous lifetime exposure of 0.1 rem per year would increase the incidence of fatal cancers in an exposed population by 2.5 percent for males and 3.4 percent for females.[69] Incidence would scale linearly with dose, in this low-dose region.[70] Thus, an average lifetime exposure of 0.33 rem per year would increase the incidence of fatal cancers by about 8 percent for males and 11 percent for females. About 21 percent of males and 18 percent of females normally die of

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[67] At a given location contaminated by cesium-137, the resulting external, whole-body dose received by a person at that location would decline over time, due to radioactive decay and weathering of the cesium-137. Thus, a person receiving 10 rem over an initial 30-year period would receive a lower dose over the subsequent 30 -year period.

[68] National Research Council, Health Effects of Exposure to Low Levels of Ionizing Radiation: BEIR V (Washington, DC: National Academy Press, 1990).

[69] Ibid, Table 4-2.

[70] The BEIR V committee assumed a linear dose-response model for cancers other than leukemia, and a model for leukemia that is effectively linear in the low-dose range. See National Research Council, 1990 (op cit), pp 171-176.

cancer.[71] In other words, in populations residing continuously at the threshold of uninhabitability (an external dose rate of 0.33 rem per year), about 2 percent of people would suffer a fatal cancer that would not otherwise occur.[72] Internal doses from contaminated food and water could cause additional cancer fatalities.

(VII-4) The increased cancer incidence described in paragraph VII-3 would apply at the boundary of the uninhabitable area. Within that area, the external dose rate from cesium-137 would exceed the threshold of 10 rem over 30 years. At some locations, the dose rate would exceed this threshold by orders of magnitude. Therefore, persons choosing to live within the uninhabitable area would experience an incidence of fatal cancers at a level higher than is set forth in paragraph VII-3.

(VII-5) For a postulated release of cesium-137 to the atmosphere, the area of uninhabitable land can be estimated from calculations done by Dr Jan Beyea. My use of these calculations is described in a report that I prepared for Orange County, North Carolina.[73] Three releases of cesium-137 are postulated here, drawn from paragraph III-5. The first release is 42 million Curies, representing the fuel that was present in the Indian Point Unit 2 pool in November 1998. The second postulated release is 31 million Curies, representing the fuel that was present in the Indian Point Unit 3 pool in November 1998. (Actual, present inventories of cesium-137 in the Unit 2 and Unit 3 pools are higher than these numbers if fuel has been added since November 1998.) The third postulated release is 3 million Curies, representing 50 percent of the cesium-137 inventory in the core of the Unit 2 or Unit 3 reactor. A release fraction of 50 percent – from the reactor core to the atmosphere – is a reasonable assumption for certain types of severe reactor accident. Higher release fractions could occur for some accident scenarios, such as a rapid-onset core melt with an open containment.

(VII-6) For typical weather conditions, a release of 42 million Curies of cesium-137 would render about 95,000 square kilometers of land uninhabitable, while a release of 31 million Curies would render about 75,000 square kilometers uninhabitable. A release of 3 million Curies would render uninhabitable about 7,500 square kilometers. For comparison, note that the area of New York state is 127,000 square kilometers. The use of a little imagination shows that a pool fire at Indian Point would be a regional and national disaster of historic proportions,

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[71] National Research Council, 1990 (op cit), Table 4-2.
[72] For males, 0.08 x 0.21 = 0.017. For females, 0.11 x 0.18 = 0.020.
[73] Thompson, February 1999 (op cit), Appendix E.

with health, environmental, economic, social and political dimensions. The long-term consequences of a severe reactor accident could also be grave.

(VII-7) The core of an operating reactor contains short-lived radioisotopes that are not present in a spent fuel pool. Notably, the core contains tellurium isotopes with half-lives of up to 3 days, and iodine isotopes with half-lives of up to 8 days. Calculations show -- for an assumed severe reactor accident designated as the SST1 accident -- that tellurium and iodine isotopes account for 70 percent of the whole-body dose received in 1 day by a person downwind of the reactor.[74] By contrast, cesium isotopes -- principally cesium-137 -- account for 66 percent of long-term radiation exposure and cancer deaths.[75]

(VII-8) A severe reactor accident will release to the atmosphere a plume that contains telluriums, iodines and other radioisotopes. The plume will travel downwind. Persons in the path of this plume could receive high radiation doses. For example, consider the plume from a PWR2 release, one of the severe accident releases examined in the NRC's 1975 Reactor Safety Study. Calculations show that the whole-body radiation dose received in 1 day by a person who is unable to shelter or escape from the plume, assuming a windspeed of 6 miles/hr and Class D atmospheric stability, will exceed 100 rem if the person is between 2.5 and 20 miles from the reactor.[76] (In this scenario, the plume passes above persons located within 2.5 miles of the reactor.) Inability to shelter or escape could arise, for example, if a person is caught in a traffic jam.

(VII-9) A guidance document published by the US Department of Health and Human Services states:[77] "Most authorities agree that observation and treatment in a specialized hospital is indicated for whole-body exposures greater than 100 rem." The same document states that the LD 50/60 (the dose that is lethal within 60 days to 50 percent of the persons exposed) is about 450 rem.[78]

(VII-10) The preceding paragraphs provide some illustrative information about the potential consequences of a reactor accident. These potential consequences

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[74] Daniel J Alpert et al, Relative Importance of Individual Elements to Reactor Accident Consequences Assuming Equal Release Fractions, NUREG/CR-4467 (Washington, DC: NRC, March 1986), page 14.
[75] Ibid.
[76] T S Margulies and J A Martin, Dose Calculations for Severe LWR Accident Scenarios, NUREG-1062 (Washington, DC: NRC, May 1984), page 36.
[77] Bernard Shleien, Preparedness and Response in Radiation Accidents (Washington, DC: US Department of Health and Human Services, August 1983), page 91.
[78] Ibid.

have been examined in great detail in a number of studies. The findings show that a variety of adverse health effects can occur, their incidence depending upon: (a) the nature of the accident; (b) weather conditions, including wind direction and speed; and (c) the ability of persons to reduce their exposure by actions such as sheltering, evacuation, respiratory protection, avoidance of contaminated food and water, and the ingestion of potassium iodide pills. Health effects can be roughly divided into two categories according to the timing of their onset. "Early" health effects are manifested over a period of days or weeks, while "late" health effects, principally cancers, are manifested years after the exposure.

VIII. OPTIONS FOR REDUCING THE VULNERABILITY OF THE INDIAN POINT STATION

(VIII-1) Populations in the vicinity of Indian Point could be protected from some of the consequences of a reactor accident by emergency response measures such as sheltering and evacuation. The effectiveness of these measures could be enhanced by increasing the scope of emergency response planning. However, emergency response measures would provide little protection from the land contamination that would arise from a reactor accident or pool fire. Also, decontamination of radioactively contaminated land and structures is difficult and expensive. Thus, the primary means of protecting populations from radiation exposure from contaminated land would be to relocate populations from the contaminated areas.

(VIII-2) In this declaration I focus on reducing the vulnerability of the Indian Point station, rather than on protecting populations from the effects of a release of radioactive material. Reducing the station's vulnerability would require the use of measures that reduce the probability and/or magnitude of a release of radioactive material as a result of a reactor accident or pool fire.

(VIII-3) Defensive measures could be taken to reduce the probability that the perpetrator of an act of malice or insanity will be able to obtain access. Measures of this kind include stronger guard forces, closer supervision of employees, vehicle barriers, no-fly zones and anti-aircraft missiles. Such measures are equally applicable to reactors and spent fuel storage facilities.

(VIII-4) Comparatively little can be done to harden the Indian Point reactors so that they are inherently more robust against acts of malice or insanity. There is, however, one option that would quickly and substantially reduce the vulnerability of the reactors. That option would be to shut down the reactors.

Their vulnerability would be reduced in two ways. First, the propensity of a reactor core to melt, if the flow of cooling water to the core is interrupted, is substantially reduced within a few hours of shutdown. Second, a reactor core's inventory of short-lived radioisotopes is substantially reduced within a few days of shutdown, thus reducing the potential incidence of early health effects and thyroid cancers in surrounding populations if an accident occurs.

(VIII-5) Practical options are available for providing more robust storage of spent fuel. In illustration, note from paragraph V-2 the design basis that was used by ASEA-Atom in developing a preliminary design for the PIUS reactor. The PIUS design basis included takeover of the plant by saboteurs, aerial bombardment, and abandonment by operators. A similar or more robust design basis could be specified for a spent fuel storage facility. Such a facility could, for example, be explicitly designed to withstand the impact of a fully-fueled Boeing 747. Meeting that requirement would not be difficult from an engineering standpoint. However, greater robustness would typically involve greater cost, and could have other implications.

(VIII-6) In examining options for storing spent fuel, the design basis for a particular storage facility is only one consideration. Other considerations include: (a) the timeframe required to implement the option; (b) whether the option involves facilities that store fuel from more than one reactor or site; (c) the extent to which the option involves transport of spent fuel between sites; (d) the closeness of storage sites to population centers; (e) security aspects of storage sites (e.g., proximity to highways, air corridors or nearby high ground); (f) the relationship of a storage option to the national strategy for long-term management of radioactive waste; and (g) cost. These and other considerations would be addressed in a credible review of options for reducing the vulnerability of the Indian Point station.

(VIII-7) Here, by way of illustration, I sketch a storage option that might be implemented at the Indian Point site. I do not recommend this option above others, but offer it as a storage option that might, upon closer examination, prove to be quickly implementable. The option would involve dry storage of spent fuel in metal casks that are robust against fire and explosion. Some of the casks that are now approved by the NRC for dry storage of spent fuel may be sufficiently robust.[79] A fuel assembly that has been discharged from the reactor comparatively recently would be stored in a low-density, open-frame rack in a

[79] NRC-approved designs for dry storage of spent fuel are listed in: US Nuclear Regulatory Commission, Information Digest, NUREG-1350 (Washington, DC: NRC, annual).

pool, and would be transferred to a cask when its heat output fell to an appropriate level. Each cask would be placed on an individual concrete pad, and would be completely surrounded by a high, earth-and-gravel berm. (The berm would be completed after placement of the cask on its pad.) The storage facility would be surrounded by a security fence, would be guarded, and would be equipped with cameras and motion detectors. The facility would not be adjacent to the Indian Point reactors.

(VIII-8) The storage option sketched in paragraph VIII-7 would be safe against a wide variety of influences, but it would not prevent a release of radioactive material in the event of a severe attack from the air. However, the magnitude of the release could be limited. For example, the release of cesium-137 as a result of an aircraft impact might be limited to the inventory in one cask. Metal casks approved by the NRC for dry storage of PWR fuel have capacities ranging from 21 assemblies to 56 assemblies.[80] At present, as described in paragraph III-2, the Indian Point Unit 2 and Unit 3 pools contain at least 917 and 672 fuel assemblies, respectively.

(VIII-9) It would be possible to construct a spent fuel storage facility at the Indian Point site that would be more robust against attack from the air than the storage option sketched in paragraph VIII-7. Meeting this objective could require placement of spent fuel storage containers in underground vaults. An option of this kind should be included in any review of options for reducing the vulnerability of the Indian Point station.

IX. SENSITIVE INFORMATION ABOUT THE VULNERABILITY OF NUCLEAR FACILITIES

(IX-1) A perpetrator of an act of malice or insanity at a nuclear facility will typically seek information about the facility's vulnerability, before committing the act. Information of this kind could improve the perpetrator's likelihood of damaging the facility, and could increase the magnitude of the radioactive release that is caused by his act. Thus, some items of information about a facility's vulnerability to acts of malice or insanity may be inappropriate for general distribution. Hereafter, such items of information are referred to as "sensitive". In the following paragraphs of Section IX, I identify a category of information that is potentially sensitive, and I sketch a process whereby sensitive information could be managed in the context of an NRC license proceeding.

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[80] NUREG-1350 (op cit), 1998 edition, Appendix G.

None of the information in this declaration is sensitive, and the declaration is appropriate for general distribution.

(IX-2) Before considering the potential need to limit the distribution of information in the context of an NRC license proceeding, it is important to consider the countervailing need for openness. There are two powerful arguments for openness about the issues that are addressed in NRC license proceedings. First, experience shows that the safety of nuclear facilities is significantly and adversely affected by a culture of secrecy. Second, secrecy about civil nuclear facilities is incompatible with democracy.

(IX-3) I have studied, observed and written about the adverse effects that a culture of secrecy has on the safety of nuclear facilities.[81] One of my findings is that the culture of secrecy in the former USSR was a major factor contributing to the occurrence of the 1986 Chernobyl reactor accident. Through direct experience, I have observed the adverse effects that a culture of secrecy has on the safety of nuclear facilities. Secrecy inhibits the development of accurate knowledge about safety problems, promotes complacency, and discourages actions that are needed to address safety problems. My direct experience has been in three contexts. In each instance, the culture of secrecy has been less intense than in the USSR, but the effects on safety have been significant and adverse. One context has been the operation of defense materials production reactors at the Savannah River site in South Carolina. The second context has been the operation of the Sellafield site in Britain. The third context has been the operation of the La Hague site in France.

(IX-4) The US nuclear industry exists to supply a commercial product -- electricity -- to the citizens of a democracy. Thus, the nuclear industry should exhibit, at a minimum, the level of openness that is expected for any industry. In addition, the operation of nuclear facilities raises significant issues related to public safety and environmental protection. Moreover, the industry's liability for damages is limited, and state governments have no power over the industry in regard to safety issues. Thus, if the operation of the nuclear industry is to be compatible with democracy, then the industry and the NRC must exhibit a level of openness that is much greater than that of other industries.

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[81] Gordon Thompson, "Science, democracy and safety: why public accountability matters", in F Barker (editor), Management of Radioactive Wastes: Issues for local authorities (London: Thomas Telford, 1998). See also: Thompson, June 1998 (op cit), Appendix E; and Thompson, May 2000 (op cit).

(IX-5) In light of the considerations addressed in paragraphs IX-3 and IX-4, any action to limit the distribution of information generated during the course of an NRC license proceeding must be regarded as a temporary measure under emergency conditions, and restriction of the distribution of information must be applied sparingly. The information that I define as "sensitive" is not commercially confidential information, classified information or safeguards information. Instead, it is information that would enter the public record during a normal licensing proceeding. The NRC Staff has stated that "discussion of the potential vulnerabilities of SFPs [spent fuel pools] to radiological sabotage is Safeguards Information (SGI)....."[82] This statement shows that the Staff uses a narrow definition of "sabotage", and does not understand the full potential for acts of malice or insanity to cause a pool fire. There are many similarities between: (a) pool fire scenarios that have been thought of as "accidents"; and (b) pool fire scenarios that are initiated by acts of malice or insanity. For example, pool fire scenarios initiated by cask drop or aircraft impact have been thought of by the Staff as "accidents", and have been examined accordingly. The Staff has never categorised information about these scenarios as safeguards information. Yet, similar scenarios could be initiated by the deliberate dropping of a cask or the deliberate impact of an aircraft.

(IX-6) If a license proceeding were to address the Riverkeeper petition that this declaration supports, the proceeding would generate a flow of information. A portion of this flow of information would relate to the potential for an act of malice or insanity to initiate a reactor accident or pool fire at Indian Point, and the consequences of such an event. All items of information that are sensitive, as defined in paragraph IX-1, would be found within this portion of the overall flow of information. Within this portion, there would be three major categories of information. The first category of information would pertain to the consequences of acts of malice or insanity. Information in this category should be generally distributed. The second category of information would pertain to the potential for acts of malice or insanity to be undertaken. For example, information about the history of terrorist events would fall into this category. Information in the second category should be generally distributed, with one possible exception. The possible exception would be detailed information about specific vulnerabilities that were exploited during past acts of malice or insanity. The third category of information would pertain to the vulnerabilities of facilities on the Indian Point site. Information in this category would be potentially sensitive. It may be appropriate to limit the distribution of some information in this category.

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[82] SECY-01-0100, 4 June 2001 (op cit), page 8.

(IX-7) Paragraph IX-6 identifies a category of information that is potentially sensitive. The category encompasses information pertaining to the vulnerabilities of nuclear facilities. However, there is already a large body of related information in the public domain. For example, there is a large, widely-available engineering literature about explosions and aircraft impacts, in general and in the context of nuclear facilities. Limiting the distribution of such literature, in the context of a license proceeding, would be a fruitless and unnecessary exercise. Instead, efforts to identify sensitive information should focus on detailed, highly-specific information. For example, a drawing showing the precise location of a vulnerable component could be sensitive information. Judgment would have to be exercised in identifying the items of information that are sensitive. Cooperation and mutual respect among the parties to a license proceeding would make the process of identifying sensitive information go more smoothly.

(IX-8) Items of information that are determined to be sensitive would be freely available to individuals who are designated by each party to a license proceeding. Sensitive information would not be generally distributed. A separate, limited-distribution record would be made of any oral or written arguments that disclose sensitive information.

X. REQUIREMENTS FOR A CREDIBLE REVIEW OF THE VULNERABILITY OF THE INDIAN POINT STATION, AND OF OPTIONS FOR REDUCING ITS VULNERABILITY

(X-1) In 1979, the NRC published a generic EIS for the handling and storage of spent fuel.[83] This EIS did not mention the potential for a pool fire. In fact, the NRC has never published an EIS that addresses the potential for a pool fire. Thus , there exists no EIS or equivalent document that provides useful guidance about the risks associated with high-density storage of spent fuel in pools. Nor is there any EIS or equivalent document that provides useful guidance about the risks associated with potential acts of malice or insanity at commercial nuclear reactors.

(X-2) At various points in this declaration, I discuss requirements for a credible review of the vulnerability of the Indian Point station, and of options for

[83] US Nuclear Regulatory Commission, Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel, NUREG-0575 (Washington, DC: NRC, August 1979).

reducing its vulnerability. Here, I summarize these requirements. These are necessary but not sufficient requirements for a credible review. I focus on requirements that address: (a) the potential for a reactor accident or pool fire; (b) the offsite consequences of a reactor accident or pool fire; and (c) options for reducing Indian Point's vulnerability.

(X-3) Paragraph IV-10 sets forth requirements for a thorough, credible study of the potential for a pool fire, and the importance of these requirements is discussed in paragraph VI-4. Such a study would be an essential part of a broader study of the vulnerability of the Indian Point station. This broader study should consider the factors mentioned in paragraph VI-13. Also, the broader study should consider potential interactions among facilities at the Indian Point site, including the potential for "cascading" scenarios that is discussed in paragraph VI-14. It is possible that sensitive information, as defined in Section IX of this declaration, would contribute to the findings of the study. In that case, the sensitive information should be cited in the study, and the sensitive information itself should be made available to authorised persons. Authorised persons would include individuals designated by the parties to any license proceeding related to the vulnerability of the Indian Point station.

(X-4) Section VII of this declaration provides a limited, illustrative discussion of the offsite consequences of a reactor accident or pool fire at Indian Point. A credible review of the site's vulnerability would provide a much more detailed examination of potential consequences. Analytic techniques suitable for such an examination are readily available. It is especially important that the review provides a thorough analysis of the long-term effects of contaminating the environment with radioactive material. Relevant effects include health, environmental, economic and social effects.

(X-5) Section VIII of this declaration provides a limited, illustrative discussion of options for reducing the vulnerability of the Indian Point station. A credible review would identify, and examine in detail, a range of options, including the spent fuel storage options sketched in paragraphs VIII-7 and VIII-9. The examination of spent fuel storage options should consider, among other factors, the issues mentioned in paragraph VIII-6. The overall review should assess the risk profiles of the various options and the present configuration of the Indian Point station, on a common basis. The risk profiles should be assessed by estimating the radioactive release potential for each option, for a common set of influences, and the attendant consequences.

XI. CONCLUSIONS

(XI-1) The Indian Point Unit 2 and Unit 3 spent fuel pools now contain at least 917 and 672 fuel assemblies, respectively; these pool inventories are equivalent to about 4.8 and 3.5 reactor cores, respectively. The Unit 2 pool now contains at least 42 million Curies of cesium-137, and the Unit 3 pool contains at least 31 million Curies. Each of the Unit 2 and Unit 3 reactors contains about 6 million Curies of cesium-137. For comparison, the 1986 Chernobyl accident released about 2.4 million Curies of cesium-137.

(XI-2) The NRC Staff concedes that a loss of water from a high-density spent fuel pool, exposing the top of the fuel racks, will lead to a fire in the pool. The Staff assumes that such a fire would release to the atmosphere 100 percent of the cesium-137 in the pool.

(XI-3) The NRC has never performed an EIS or equivalent document that addresses the potential for a pool fire. In defense of this omission, the NRC has asserted that a pool fire is a "remote and speculative" event.

(XI-4) The NRC Staff has conducted a number of studies related to the potential for a pool fire. There are numerous deficiencies and omissions in these studies. Notably, the Staff's studies have neglected the potential for a pool fire to be caused by an act of malice or insanity. Similarly, neither the NRC nor any of its licensees has conducted a credible study of the potential for a reactor accident to be caused by an act of malice or insanity.

(XI-5) The occurrence of a highly-destructive act of malice or insanity at a nuclear power station has been foreseeable for many years, and has been foreseen. The terrorist attacks of 11 September 2001 on the World Trade Center and the Pentagon provide additional information. These attacks demonstrate conclusively that a highly-destructive act of malice or insanity at the Indian Point site is not a remote and speculative event.

(XI-6) Available information indicates that acts of malice or insanity at the Indian Point site, including but not limited to the deliberate impact of a large aircraft, could initiate a reactor accident and/or pool fire at Unit 2 and/or Unit 3.

(XI-7) The offsite consequences of a pool fire at Indian Point Unit 2 could include the rendering uninhabitable of a land area of about 95,000 square kilometers, and a pool fire at Unit 3 could render uninhabitable a land area of about 75,000 square kilometers. For comparison, the area of New York state is 127,000 square kilometers.

(XI-8) Options for reducing the vulnerability of the Indian Point station are available. The vulnerability of the Unit 2 and Unit 3 reactors could be quickly and substantially reduced by shutting them down. Options are available for storing spent fuel that would be much more robust than the Indian Point Unit 2 and Unit 3 pools in terms of their ability to withstand accidents or acts of malice or insanity.

(XI-9) The vulnerability of the Indian Point station, and options for reducing its vulnerability, should be systematically reviewed. Some necessary, but not sufficent, requirements for such a review are set forth in Section X of this declaration.

I declare, under penalty of perjury, that the foregoing facts are true and correct to the best of my knowledge and belief, and that the opinions expressed above are based on my best professional judgment.

Executed on 7 December 2001.

_____
Gordon Thompson

December 20, 2001

Mr. Alex Matthiessen
Executive Director
Riverkeeper, Inc.
25 Wing & Wing
Garrison, NY 10524

Dear Mr. Matthiessen:

On November 8, 2001, you filed a Petition on behalf of Riverkeeper, Inc., et.al., with the U.S. Nuclear Regulatory Commission (NRC). You requested that the NRC: (1) order the licensee to suspend operations, revoke the operating license, or adopt other measures resulting in a temporary shutdown of the Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and 3); (2) order the licensee to conduct a full review of the facility's vulnerabilities, security measures and evacuation plans; (3) require the licensee to provide information documenting the existing and readily attainable security measures which protect the IP facility against land, water, and airborne terrorist attacks; (4) immediately modify the IP2 and 3 operating licenses to mandate certain specified security measures sufficient to protect the facility; and (5) order the revision of the licensee's Emergency Response Plan and Westchester County's Radiological Emergency Preparedness Plan (REPP) to account for possible terrorist attacks and prepare a comprehensive response to multiple attacks in the region which may impair the efficient evacuation of the area. In addition, you state that if, after conducting a full review of the facility's vulnerabilities, security measures, and evacuation plans, the NRC finds that it cannot sufficiently ensure the security of the IP facility against terrorist threats, the NRC should take prompt action to permanently retire the facility. Further, separately from the above issues, you request that the NRC order the licensee to undertake the immediate conversion of the current spent fuel storage technology from a water-cooled system to a dry cask system.

As the basis for your requests, you stated that (1) the IP facility is a plausible target of future terrorist actions, (2) actual threats against nuclear power plants have been documented, (3) IP is currently vulnerable to catastrophic terrorist attack, (4) a terrorist attack on IP2 and 3 would have significant public health, environmental, and economic impacts, and (5) the Westchester County's REPP is inadequate because it is based on erroneous assumptions.

The NRC has, in effect, partially granted your request in item for above. On September 11, 2001, the NRC quickly took action to enhance security at all nuclear facilities, including IP2 and 3. Immediately after the attacks, we advised all nuclear power plants to go to the highest level of security, which they promptly did. These facilities have remained at an appropriate and heightened security level since that time. The NRC continues to work with other Federal agencies, and is monitoring all relevant information it receives on security matters at nuclear facilities. We are prepared to make immediate adjustments as necessary to continue to assure adequate protection of the public.

We also understand and recognize that many of our fellow citizens are concerned that a U.S. nuclear facility could be a target of future terrorist action. In view of the recent unprecedented events, Chairman Richard A. Meserve, with the full support of the Commission, has directed the staff to undertake a top-to-bottom review of our security regulations and procedures, including the basic assumptions of our current programs. This review involves U.S. national security organizations, and is part of a broader review being undertaken by the Federal Government. As you may be aware, the State of New York's Office of Public Security worked with various Federal and State agencies, including the Federal Bureau of Investigation (FBI), to assess the long-term security needs at IP. As a result, an FBI report was prepared that provided several recommendations to enhance security. Many of the measures suggested by the FBI have been implemented by the licensee and others are currently under advisement.

Although we cannot rule out the possibility of future terrorist activity directed at one of our licensee sites before implementing any potential enhancements to our safeguards programs, we believe that these facilities can continue to operate safely. Nuclear power plants are inherently robust. Their design is based on defense-in-depth principles, and includes many features to protect public health and safety. For example, reinforced containment buildings and redundant safety systems would help assist trained operators in preventing or limiting the release of radioactive material in the unlikely event of a terrorist attack. Therefore, in light of the facility's defense-in-depth, the heightened security measures implemented in response to the events of September 11th, and the NRC's ongoing reevaluation of its safeguards regulations and programs, we do not consider the immediate closure of IP2 and 3 to be necessary to provide adequate protection for the public's health and safety.

Under normal circumstances, we closely follow Management Directive (MD) 8.11, "Review Process for 10 CFR 2.206 Petitions," when reviewing requests for enforcement action; however, since your letter involves sensitive security issues, we are deferring application of certain public aspects of the MD 8.11 process pending further developments of our security review. Ordinarily, we give individuals who have petitioned the NRC an opportunity to provide additional information to the Office of Nuclear Reactor Regulation Petition Review Board (PRB) before determining whether a request is to be reviewed as a petition pursuant to 10 CFR 2.206. However, since your request is already being treated as a 2.206 Petition, and because of the sensitive nature of this information, we have determined that a public meeting or telephone conference with the PRB is not necessary or appropriate at this time.

We will take actions on your request within a reasonable time, using the 10 CFR 2.206 Petition process where possible. Mr. Patrick Milano has been assigned as the Petition Manager for your request. Mr. Milano may be reached at 301-415-1457 (email: pdm@nrc.gov). Your Petition is presently being reviewed by members of the NRC's Office of Nuclear Reactor

Regulation. Enclosed for your information are: (1) a copy of the notice that is being filed with the Office of the Federal Register for publication; (2) a copy of MD 8.11; and (3) NRC brochure NUREG/BR-0200, Revision 4, "Public Petition Process," prepared by the NRC Office of Public Affairs.

Sincerely,

/RA/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 286

Enclosures:
1. *Federal Register* Notice
2. Management Directive 8.11
3. NUREG/BR-0200

cc w/o encls: See next page

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7590-01-P

U.S. NUCLEAR REGULATORY COMMISSION

THE UNPLUG SALEM CAMPAIGN

THE NATIONAL WHISTLEBLOWER CENTER AND MR. RANDY ROBARGE

RIVERKEEPER, INC., ET AL

RECEIPT OF REQUESTS FOR ACTION UNDER 10 CFR 2.206

Notice is hereby given that by the following three petitions, the Nuclear Regulatory Commission (NRC) was requested to take immediate corrective actions to protect the public against the possibility of terrorists seizing control of a large commercial jetliner and crashing into a nuclear power plant in the United States.

1. From Mr. Norm Cohen, on behalf of the UNPLUG Salem Campaign, dated September 17, 2001.
2. From Mr. Michael D. Kohn, on behalf of the National Whistleblower Center and Randy Robarge, dated October 24, 2001.
3. From Messrs. Alex Matthiessen, and Karl Coplan, on behalf of the Riverkeeper, Inc., et al, dated November 8, 2001.

The petitioners requested that the NRC staff take certain specified compensatory measures, to protect the public and environment from the catastrophic impact of a terrorist attack on a nuclear power plant or a spent fuel pool.

These requests are being treated pursuant to 10 CFR 2.206 of the Commission's regulations. These requests have been referred to the Director of the Office of Nuclear Reactor Regulation. As provided by Section 2.206, appropriate action will be taken on these petitions within a reasonable time.

February 11, 2002

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Director, Office of Nuclear Reactor Regulation

In the Matter of	)	
	)	
ENTERGY NUCLEAR INDIAN POINT 2, LLC,	)	Docket Nos. 50-247
ENTERGY NUCLEAR INDIAN POINT 3 LLC,	)	and 50-286
and ENTERGY NUCLEAR OPERATIONS, INC.	)	(License Nos. DPR-26
	)	and DPR-64)
	)	

LICENSEES' RESPONSE TO RIVERKEEPER, INC.'S SECTION 2.206 REQUEST FOR EMERGENCY SHUTDOWN OF INDIAN POINT UNITS 2 AND 3

I. INTRODUCTION

Licensees Entergy Nuclear Indian Point 2, LLC, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Operations, Inc. (collectively, "Entergy") file this response in opposition to the request ("Request") filed by Riverkeeper, Inc., et al. (collectively, "Petitioners"), pursuant to 10 C.F.R. §§2.206 and 2.202, for the emergency shutdown of the Indian Point Units 2 and 3 nuclear power plants (collectively, the "Indian Point Units" or the "Units").[1] Based on the tragic events of September 11, 2001, Petitioners seek immediate shutdown of the Indian Point Units, among other measures.

[1] Entergy Nuclear Indian Point 2, LLC and Entergy Nuclear Indian Point 3, LLC are the respective owners of Indian Point 2 and Indian Point 3. Entergy Nuclear Operations, Inc. (hereinafter "ENO") is the licensed operator of both units.

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The extreme relief requested by Petitioners is based on speculative concerns, lacks an adequate legal foundation, and is factually unsupported. Therefore, Entergy submits that it must be denied.[2]

As more fully discussed below, Petitioners neither allege nor cite a legal basis for their Request. In particular, Petitioners advance no claim that the design or operation of the Indian Point Units violates the Atomic Energy Act or applicable Nuclear Regulatory Commission ("NRC" or the "Commission") regulations. Rather, Petitioners mount an impermissible collateral attack on the NRC regulations relating to terrorist acts, as they apply to the Indian Point Units.

Petitioners ignore that the NRC is conducting a full review of security requirements for nuclear reactors in light of the September 11 events. It is settled that generic issues, such as plant protection against terrorist acts, are best addressed through rulemaking; indeed, NRC precedent clearly prohibits granting a §2.206 petition where the Commission is already addressing on a generic basis the issues which are the subject of the petition. Since there is an ongoing generic review by the NRC of nuclear power plant security, there should not be concern that, if the Request is denied, Petitioners will be left without an adequate forum. In a generic rulemaking proceeding, Petitioners would be afforded a full opportunity to participate in the rulemaking, consistent with applicable NRC regulations. *See* 10 C.F.R. §2.805(a).

Moreover, Petitioners' Request lacks a credible factual basis. Petitioners' Requests rests solely on postulated "terrorist attacks" directed against the Indian Point Units, which Petitioners contend may lead to radioactive releases from the Units. The

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[2] The portion of the Request that seeks the immediate shutdown of the Indian Point Units has already been denied by the NRC Staff. See letter from Samuel J. Collins (NRC) to Alex Matthiessen (Riverkeeper, Inc.) dated December 20, 2001 ("Collins letter").

proffered scenarios are based on erroneous assumptions or are so improbable as to lack credibility. Further, the Request ignores the heightened state of alert that currently prevails in the U.S. society at large, and at nuclear power plants in particular. It also ignores the considerable measures already taken at the Indian Point Units to reduce the reasonably foreseeable threat of malevolent acts against the Units. Thus, the Request does not provide a factual predicate on which a potential shutdown of the Indian Point Units could be based.

In short, no legal basis exists for granting the extraordinary relief Petitioners seek, and they have come forward with no credible factual information supporting NRC action against Indian Point. Their Request, therefore, must be denied.

II. FACTUAL BACKGROUND

A. THE INDIAN POINT UNITS ARE IN FULL COMPLIANCE WITH CURRENT NRC SECURITY REQUIREMENTS

The NRC has established stringent physical security requirements for commercial nuclear plants, with which the Indian Point Units must comply. Under 10 C.F.R. § 73.55, a security organization and plant physical protection systems must be in place that provides adequate protection against attacks from external armed groups and internal saboteurs, as defined by the "design basis threat" set forth in 10 CFR §73.1(a)(1). The plant security measures must be able to prevent unauthorized access of personnel, vehicles, and materials; ensure only authorized activities are conducted; permit only authorized handling of nuclear material; and detect and respond to unauthorized penetrations. The perimeter must be monitored both visually and electronically with electronic alarms sounding at two independent continuously staffed stations. Entry points must be guarded and monitored and access must be strictly controlled. All plants must have armed response forces whose qualifications and tactical training are dictated by 10

C.F.R. Part 73, Appendix B. Each armed responder must be capable of maintaining continuous communication with each of the continuously staffed alarm stations.

A written Safeguards Contingency Plan must be developed and maintained by nuclear plants in accordance with 10 C.F.R. Part 73, Appendix C, identifying a predetermined set of threat response actions, their means of implementation, and the personnel responsible for responding to the threats.[3] Further, nuclear plants are required to establish and document a working liaison with local law enforcement authorities, whom they can summon for assistance in the event of an attack.

Threats at nuclear plants are to be countered by an armed tactical force permanently stationed at the plant, whose mission is to quickly ascertain the threat's existence, assess its magnitude, and interpose itself between the threat and specific key plant areas. The capability of security response forces and systems to defend against threats is tested in live exercises monitored by the NRC using mock attack forces. *See* NRC Inspection Manual, Inspection Procedure 81110, Operational Safeguards Response Evaluation ("OSRE") (July 1, 1997). If weaknesses are identified, the plant must institute additional defensive countermeasures.

The Indian Point Units have in place formal, documented Physical Security Protection Programs and Safeguards Contingency Plans. These documents have been reviewed and approved by the NRC.[4] The existence and implementation of these plans and their approval by the NRC assure that appropriate levels of security exist at Indian Point, in accordance with current regulatory requirements.

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[3] The contents of the Safeguards Contingency Plan are confidential information protected from public disclosure. 10 C.F.R. §73.21(b)(1)(viii).

[4] *See, e.g.*, April 10, 2001 NRC Supplemental Inspection Report 05000247/2001-002 (Indian Point Unit 2).

B. HEIGHTENED SECURITY MEASURES HAVE BEEN IMPLEMENTED BY THE UNITED STATES, THE NRC AND THE NUCLEAR INDUSTRY SINCE THE SEPTEMBER 11, 2001 TERRORIST ATTACKS

Immediately following the attacks of September 11, 2001, the NRC issued a threat advisory asking all nuclear plants to implement a heightened state of alert. In response, as reported by NRC Chairman Richard Meserve, nuclear plants have augmented security forces and patrols, increased coordination with law enforcement and military authorities, imposed additional site access limitations for personnel and vehicles, and taken other short-term and long-term actions to strengthen plant capability to respond to terrorist attacks, all as provided by an October 6, 2001 "safeguards advisory" issued by the NRC.[5]

The NRC has described its own post-September 11 initiatives with regard to the security of nuclear facilities as follows:

> In response to the September 11, 2001 terrorist attacks, the Commission has taken a number of actions to ensure the security of NRC-licensed facilities and materials, including activation and staffing the NRC Operations Center on a 24-hour-a-day basis. Immediately following the attacks, the NRC advised nuclear power plant licensees and fuel facilities to go to the highest level of security, and all promptly did so. In addition, the Commission has had continuous and close coordination with the Federal Bureau of Investigation, other intelligence and law enforcement agencies, the Office of Homeland Security, NRC licensees, and military, state and local authorities. The Commission has issued security advisories to licensees to update them on the available threat information and to recommend additional security measures. The Commission continues to monitor the situation, and is prepared to make any adjustments to security measures for NRC-licensed activities as may be deemed appropriate.

Private Fuel Storage L.L.C. (Independent Spent Fuel Storage Installation), CLI-01-26, 54 NRC __, ___ (December 28, 2001), slip op. at 2-3. *See also,* Collins letter, *supra.*

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[5] *See* Letter from NRC Chairman Richard Meserve to Senator James Jeffords (December 17, 2001).

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As noted by the NRC, since September 11, 2001 the Indian Point Nuclear Power Plant has assumed a heightened level of security based on a series of threat advisories issued by the NRC.[6] The steps recommended by the NRC included increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.[7] These steps have been implemented and are being maintained, since the NRC also recommended that the heightened level of security be kept in effect due to the uncertainty about the possibility of additional terrorist attacks..

Further, since the September 11 attacks, our nation has greatly increased its vigilance on all fronts against terrorism. In particular, airport security has been greatly enhanced and more stringent inspections and weapon detection procedures have been instituted. Law enforcement is on an extremely high state of alert and major efforts have been undertaken to track down potential terrorists and thwart any future attacks. This effort goes far beyond our borders. Countries around the world have also joined the fight against terrorism and taken measures analogous to those in effect in the United States.

The implementation of heightened facility, local, state, federal and international security provisions reduces the likelihood of occurrence of a terrorist act against such facilities, including the Indian Point Units. Thus, whatever level of risk remains of an act of sabotage or terrorism against Indian Point, that level is indisputably lower than it was before September 11, 2001.

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6 Letter from G. Scott Barber (NRC) to J. Barrett (Entergy) dated December 31, 2001, enclosing NRC Inspection Report 50/286/01-09) at 1.

7 Id. at 1-2.

C. THERE IS AN ONGOING, COMPREHENSIVE NRC GENERIC REVIEW OF NUCLEAR PLANT SECURITY REQUIREMENTS

The Commission has announced that it is undertaking a review of nuclear power plant security requirements in light of the September 11 events:

> The Commission believes that its response to these unsettling events has been expeditious and that the current safeguards and physical security programs provide for a very high level of security at NRC-licensed facilities. However, in the aftermath of the terrorist attacks and the continuing uncertainty about future terrorist intentions, we have commenced a thorough review of our safeguards and physical security programs, from top to bottom, including those applicable to independent spent fuel storage installations. The review will involve a comprehensive examination of the programs' basic underlying assumptions.
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> Historically, the NRC has drawn a distinction between requiring its licensees to defend their facilities against sabotage and requiring them to protect against attacks and destructive acts by enemies of the United States. Even NRC-licensed facilities that are required to meet the most stringent security requirements (because the potential consequences of sabotage are greatest) are not required to protect against enemies of the United States. . . . The top-to-bottom review of our physical protection regulations will consider these distinctions, which have been underlying principles of the Commission's regulations in this area, and apply them as appropriate. The consideration of any adjustments to licensee, federal, state, and local response capabilities is being conducted in consultation with the appropriate authorities.

Private Fuel Storage L.L.C., *supra,* slip op. at 3-4. These statements by the Commission denote a clear intention on the part of the agency to review, in a thorough and comprehensive manner, the existing security arrangements at nuclear power plants and other nuclear facilities.

III. THERE IS NO LEGAL BASIS FOR THE RELIEF REQUESTED

A. SECTIONS 2.206 AND 2.202 PROVIDE NO BASIS FOR THE REQUESTED SHUTDOWN OF THE INDIAN POINT UNITS

Petitioners seek immediate shutdown of the Indian Point Units pending a "full review" of the Units' "vulnerabilities, security measures and evacuation plans." Request

at 1.[8] Petitioners ground their requested relief in 10 C.F.R. §§2.206 and 2.202.[9] However, as discussed below, the Request fails to satisfy §§2.206 and 2.202 and must be rejected.

The institution of proceedings pursuant to §2.206 is appropriate only where substantial health and safety issues have been raised. *See* Consolidated Edison Co. of New York (Indian Point, Units 1, 2, and 3), CLI-75-8, 2 NRC 173, 175-76 (1975); Northern Indiana Public Service Co. (Bailly Generating Station, Nuclear-1), CLI-78-7, 7 NRC 429, 433 (1978), aff'd, Porter County Chapter v. NRC, 606 F.2d 1363 (D.C. Cir. 1979); Yankee Atomic Electric Co. (Yankee Nuclear Power Station), CLI-96-6, 43 NRC 123, 128 (1996). Compliance with NRC requirements provides reasonable assurance of adequate protection of the public health and safety. *See, e.g.,* 10 C.F.R. Part 50 Specific Exemptions; Clarification of Standards, 50 Fed. Reg. 50,764, 50,768, 50,771 (1985); Thermo-Lag Fire Barrier Material, DD-96-3, 43 NRC 183, 195 (1996); Ohio Citizens for Responsible Energy, DPRM-88-4, 28 NRC 411, 415 (1988). Where, as here, there is no allegation of violation of NRC regulations, there can be no basis for asserting that a

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[8] The Request also seeks that Entergy be required to provide information on the existing and readily attainable security measures which provide the Indian Point facility protection against terrorist attacks; that the operating licenses for Indian Point Units 2 and 3 be amended to incorporate specified security provisions; that the Emergency Response Plans of Entergy and Westchester County be revised to "account and prepare for possible terrorist attacks"; that the Commission promptly order the retirement of the Indian Point facility if "the NRC cannot sufficiently ensure the security of the Indian Point facility against terrorist threats"; and that the Commission order Entergy "to undertake the immediate conversion of the current spent fuel storage technology from a water cooled system to a dry cask system in a bunkered structure." Id. at 1-2. As will be discussed below, there is no legal basis for any of these requested actions, nor are they warranted by current conditions at Indian Point or elsewhere.

[9] 10 CFR §2.206(a) reads, in relevant part: "Any person may file a request to institute a proceeding pursuant to §2.202 to modify, suspend or revoke a license, or for any other action as may be proper ... The Request must specify the action requested and set forth the facts that constitute the basis for the Request." 10 C.F.R. §2.202(a) states, in relevant part: "The Commission may institute a proceeding to modify, suspend, or revoke a license or to take such other action as may be proper ... [I]f the Commission finds that the public health, safety, or interest so requires or that the violation or conduct causing the violation is willful, the order may provide, for state reasons, that the proposed action be immediately effective pending further order."

substantial health and safety issue exists that warrants action under §2.206.[10] *See, e.g.*, Connecticut Yankee Atomic Power Co. (Haddam Neck Plant), DD-01-2, 53 NRC 333, 337 (2001) ("NRC will not take action to suspend or revoke [licensee's] license as no violations [of NRC regulations] occurred").[11]

B. THE OTHER RELIEF REQUESTED BY PETITIONERS AMOUNTS TO AN IMPERMISSIBLE COLLATERAL ATTACK ON COMMISSION REGULATIONS

Petitioners seek to impose on Entergy the obligation to address enemy or terrorist attacks in excess of the requirements of the Atomic Energy Act and the Commission's implementing regulations. In so doing, the Request is an impermissible collateral challenge to NRC regulations and, on that basis alone, must be rejected. Dominion Nuclear Connecticut Inc. (Millstone Nuclear Power Station, Units 2 and 3), CLI-01-24, 54 NRC ___, ___ (Dec. 5, 2001), slip. op. at 21; Duke Energy Corp. (Oconee Nuclear Station, Units 1, 2, and 3), CLI-99-11, 49 NRC 328, 334 (1999); Public Service Co. of New Hampshire (Seabrook Station, Units 1 and 2), CLI-89-8, 29 NRC 399, 416-17 (1989); Long Island Lighting Co. (Shoreham Nuclear Power Station, Unit 1), CLI-87-12, 26 NRC 383, 395 (1987). The proper vehicle for challenging NRC regulations is a rulemaking petition. *See, e.g.*, Metropolitan Edison Company (Three Mile Island Nuclear

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[10] As further discussed below, the proper vehicle for raising a potential health and safety issue not contemplated by existing NRC regulations is the filing of a rulemaking petition under 10 C.F.R. §2.802, not an action against an individual licensee under 10 CFR §2.206 to enforce existing regulations.

[11] Nor is there credible evidence that requires, as Petitioners seek, the *immediate* shutdown of the Indian Point Units. An order to modify, suspend or revoke a license may be immediately effective, in accordance with 10 C.F.R. §2.202(a)(5), if NRC "finds that the public health, safety, or interest so require ... ." Such summary action is a drastic measure, not to be granted except in exceptional circumstances, where it is "warranted by compelling safety considerations." Consumers Power Co. (Midland Plant, Units 1 and 2), CLI-73-38, 6 AEC 1082, 1083 (1973). To meet this standard, Petitioners must provide credible evidence that a threat exists of sufficient magnitude to warrant summary action. Petitioners have provided no credible evidence that an enemy or terrorist attack against the Indian Point Units is probable, let alone imminent. Thus, no compelling considerations exist that would justify the immediate relief sought in the Request, as has already been ruled by the NRC staff. See Collins letter, *supra*.

Station, Unit No. 1), 11 NRC 674, 675 (1980). Indeed, as further discussed below, an appropriate disposition of a §2.206 petition that seeks a change in NRC regulations is to treat it as a petition for rulemaking. NRC Office of Nuclear Reactor Regulation, Directive 8.11, "Review Process for 10 C.F.R. 2.206 Petitions" (October 25, 2000) ("Directive 8.11") at 12; 10 C.F.R. §2.802(a) ("Any interested person may petition the Commission to issue, amend or rescind any regulation.").

1. The Requested Actions Are Not Required by the Regulations, the Licensing Basis or the Licensing Commitments

Petitioners' first request for relief seeks immediate temporary suspension of operations at Indian Point, and asks that NRC conduct a "full review" of the Units' purported vulnerabilities, security measures and evacuation plans, among other things. Request at 1. The Petitioners advance five theories for their Request, as follows:

a. While operational, Indian Point is unnecessarily vulnerable to risks from takeover of or damage to control rooms;

b. The reactor walls were not designed to withstand the "accidental or intentional crash of fuel-laden jetliners";

c. The operating facility has multiple vulnerable points in security;

d. NRC and Indian Point personnel "confront dual challenges when ensuring security at an operational facility" and as a result the resources of "both the agency and the licensee are stretched thin by this double-taking"; and

e. "Shutting down the Indian Point Reactors creates a more secure environment. . . . Security of spent fuel has never been demonstrated at Indian Point."

Request at 6-8.

The "vulnerabilities" alleged by Petitioners, if accepted as valid and deemed to require being addressed, would impose on Entergy obligations exceeding those set by the licenses for the Indian Point Units and NRC regulations. Petitioners' Request thus seeks

to alter existing applicable security and emergency planning requirements for Indian Point and require Entergy to provide protection against a terrorist "act of war" against the Indian Point Units.

In so doing, Petitioners ignore applicable law and precedent holding that NRC licensees are not required to provide specific protection against attacks by enemies of the United States:

> [A license applicant] is not required to provide for design features or other measures for the specific purpose of protection against the effects of ... *attacks and destructive acts, including sabotage, directed against the facility by an enemy of the United States, whether a foreign government or other person ... .*

10 C.F.R. §50.13 (emphasis supplied).[12] *See also* Florida Power & Light Co. (Turkey Point Nuclear Generators Units 3 and 4), 4 AEC 9, 13 (1967), aff'd, Siegel v. AEC, 400 F.2d 778 (D.C. Cir. 1968) ("It bears emphasis that... neither [NRC] regulations nor [NRC] decisions indicate any requirement that an applicant provide for special design features or other measures for the specific purpose of protection against the effects of enemy attacks and destructive acts.").

The Commission has enunciated sound policy reasons for this rule:

> It would appear manifest, as an initial proposition, that the protection of the United States against hostile enemy acts is a responsibility of the nation's defense establishment and of the various agencies of our Government having internal security functions.

Turkey Point, supra, 4 AEC at 9. The Commission has further explained:

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[12] "Attacks and destructive acts" are those above and beyond the threats against which the reactor's physical protection system must defend under the Commission's specific security requirements in 10 C.F.R. Part 73. Commonwealth Edison Co. (Braidwood Nuclear Power Station, Units 1 and 2), LBP-85-27, 22 NRC 126, 137-138 (1985).

> One factor underlying [the Commission's] practice in this connection has been a recognition that [facility] design features to protect against the full range of the modern arsenal of weapons are simply not practicable and that the defense and internal security capabilities of this country constitute, of necessity, the basic "safeguards" as respects possible hostile acts by an enemy of the United States.
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> The circumstances which compel [the Commission's] recognition are not, of course, unique as regards a nuclear facility; they apply also to other structures which play vital roles within our complex industrial economy. The risk of enemy attack or sabotage against such structures, like the risk of all other hostile acts which might be directed against this country, is a risk that is shared by the nation as a whole. This principle, we believe, is rooted in our political history and we find no Congressional indication that nuclear facilities are to be treated differently in the subject regard.

Id. In Consolidated Edison Co. of New York, Inc. (Indian Point Station, Unit No. 2), ALAB-202, 7 AEC 825, 829-30 (1974), the Appeal Board held that an attack by "an armed band of trained saboteurs" – the type of scenarios Petitioners postulate – would constitute an enemy attack under 10 C.F.R. §50.13, regardless of the actual nature or allegiance of the attackers. The Appeal Board further held that, "an applicant should be entitled to rely on settled and traditional governmental assistance in handling [such an] attack." Id. at 830.

In another Commission licensing proceeding, it was held that a scenario involving terrorists commandeering a large aircraft and flying it into the containment structure of a nuclear power plant falls squarely within the prohibition in 10 C.F.R. §50.13. Carolina Power & Light Co. (Shearon Harris Nuclear Power Plant, Units 1 and 2), LBP-82-119A, 16 NRC 2069, 2098 (1982). The Atomic Safety and Licensing Board in the Harris proceeding confirmed that the security of nuclear power plants against hostile attacks of the nature the Request contemplates is the primary responsibility of the nation's defense and internal security establishments, not the NRC.

More recently, during the 1991 Gulf War, the Director of the Office of Nuclear Material Safety and Safeguards rejected a request for emergency action analogous to the one made by Petitioners here. *See* All Nuclear Facilities, DD-91-1, 33 NRC 53 (1991). In that case, the NRC was asked to require that "existing licensee contingency plans against truck bombs be put into effect immediately and that immediately thereafter, the NRC undertake an evaluation of the adequacy of the plans and require any such improvements as it deems necessary." 33 NRC at 54. The Director denied the §2.206 request because the requested action went beyond then existing requirements and the NRC, while reviewing the postulated threat, had not determined that any credible threat of terrorist action against NRC-licensed facilities warranted implementation of the actions sought by the petitioners. Id. The same result should be reached here.[13]

Finally, only a few weeks ago, the Commission reiterated its position that nuclear power plants are not required to be protected against terrorist acts by enemies of the United States:

> Even NRC-licensed facilities that are required to meet the most stringent security requirements (because the potential consequences of sabotage are greatest) are not required to protect against enemies of the United States. For example, reactor licensees are required to protect against a prescriptive list of possible threats, referred to collectively as the "design basis threat." However, our regulations stipulate that power reactors are not required to be designed or to provide other measures to counteract destructive acts by "enemies of the United States." The basis for this distinction is that the national defense establishment and various agencies having internal security functions have the responsibility to address this contingency, and that requiring reactor design features to protect against the full range of the modern arsenal of weapons is simply not practical.

Private Fuel Storage L.L.C., *supra,* slip op. at 3-4.

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[13] Subsequently, protection against truck bombs was added to the design basis threat requirements. *See* 59 Fed. Reg. 38889 (1994).

In Siegel v. AEC, the U.S. Court of Appeals for the District of Columbia Circuit explained the underpinnings of the policy illustrated by the above decisions. The court cited the following factors as providing the rationale for excluding consideration of potential military attacks by foreign enemies in establishing the design basis for nuclear reactors:

> (1) The impracticability, particularly in the case of civilian industry of anticipating accurately the nature of enemy attack and designing defenses against it, (2) the settled tradition of looking to the military to deal with this problem and the consequent sharing of its burdens by all citizens, and (3) the unavailability, through security classification and otherwise, of relevant information and the undesirability of ventilating what is available in public proceedings.

Siegel v. AEC, supra, 400 F.2d at 750.

This policy does not leave a gap in the safety of nuclear reactors. Existing NRC regulations (at 10 C.F.R. Part 73) already address the "physical protection of nuclear plants and facilities," including reasonably foreseeable and addressable industrial security risks, as distinct from the threats, attacks or sabotage by enemies of the United States whose consideration is precluded by 10 C.F.R. §50.13.

NRC regulations require that each licensee "provide physical protection at a fixed site ... where licensed activities are conducted, against radiological sabotage ... in accordance with the applicable sections of this Part ... ." 10 C.F.R. §73.40. According to 10 C.F.R. §73.55(a), the physical protection requirements for nuclear power reactors include a specified "design basis" threat ("DBT"), which must address "radiological sabotage," including:

> (i) A determined violent external assault, attack by stealth, or deceptive actions, of several persons with the following attributes, assistance and equipment:
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> (A) Well-trained (including military training and skills) and

dedicated individuals,

(B) inside assistance which may include a knowledgeable individual who attempts to participate in a passive role (*e.g.*, provide information), an active role (*e.g.*, facilitate entrance and exit, disable alarms and communications, participate in violent attack), or both,

(C) suitable weapons, up to and including hand-held automatic weapons, equipped with silencers and having effective long range accuracy,

(D) hand-carried equipment, including incapacitating agents and explosives for use as tools of entry or for otherwise destroying reactor, facility, transporter, or container integrity or features of the safeguards system, and

(E) a four-wheel drive land vehicle used for transporting personnel and their hand-carried equipment to the proximity of vital areas, and

(ii) An internal threat of an insider, including an employee (in any position), and

(iii) A four-wheel drive land vehicle bomb.

10 C.F.R. §73.1(a)(1). Pursuant to the DBT definition, a station's security plan must be designed to cope with a violent external assault by a discrete number of persons equipped with light, portable weapons, as distinct from the military-style attacks to which the prohibition in 10 C.F.R. §50.13 applies. *See* Carolina Power & Light Company (Shearon Harris Nuclear Power Plant, LBP-82-119A, 16 NRC 2069, 2098 (1982). While providing protection against the DBT, "[licensees] are not required to design against such things as artillery bombardments, missiles with nuclear warheads, *or kamikaze dives by large airplanes*, despite the fact that such attacks would damage and may destroy a commercial reactor." Id. (emphasis supplied).

The Indian Point Units have security plans in place that meet the requirements of 10 C.F.R. §73.55 and are capable of responding to a design basis threat. Entergy is in

compliance with those plans and, therefore, with current NRC regulations. Accordingly, there is no basis for the relief sought by Petitioners.

2. The Request that the NRC Require Entergy to Provide Information Documenting that Existing Security Measures Are Sufficient against Threats of Terrorist Attacks Seeks Actions Not Required by the Regulations, the Indian Point Units' Licensing Basis, or their Licensing Commitments

Petitioners' second request for relief asks that Entergy supply information to the NRC showing that it has provided the Indian Point facility "with protection against land, water, and airborne terrorist attacks." Request at 21. Moreover, Petitioners assert that "the design-basis threat for Indian Point did not consider the possibility of an intentional terrorist attack from the air or water, or a suicide attack from any front." Id.

Again, Petitioners seek to require Entergy to show it meets a beyond design basis threat, thus their request constitutes an impermissible attack on the Commission's regulations. A four-wheel drive land vehicle bomb is specifically provided as part of the design basis threat, but the regulations presume that only land-based vehicles are used in the attack. This was noted by the Commission in adopting 10 CFR §73.51, which sets forth the specific requirements for the physical protection of stored spent nuclear fuel at Independent Spent Fuel Storage Installations. There, the Commission responded to a comment requesting that §73.51 address both land based and airborne vehicle attacks by pointing out that 10 CFR §73.1 did not include an airborne vehicle attack as part of the design basis for nuclear power plants:

> Inclusion of an airborne vehicle was assessed for possible inclusion into the protection goal for this rule. However, protection against this type of threat has not yet been determined appropriate at sites with greater potential consequences than spent fuel storage installations.

"Final Rule: Physical Protection for Spent Nuclear Fuel and High-Level Radioactive Waste" 63 Fed. Reg. 26955, 26956 (1998).[14]

On the other hand, to the extent that Petitioners are seeking that Entergy confirm to the NRC that it meets current security requirements, such an action is unnecessary. As discussed above, the Indian Point Nuclear Power Station has an approved security plan in place that meets the requirements of 10 C.F.R. §73.55.[15] Entergy is in compliance with present NRC regulations, and there is no basis for the relief sought by Petitioners.

3. The Request that the NRC Create a System to Protect Against Air and Water Based Attacks Seeks Actions that the NRC cannot Take and Which Are not Required by the Regulations

Petitioners' third request for relief asks the Commission to create a permanent no-fly zone around the Units, to establish a defense and security system to enforce the no-fly zone, and to establish a defense and security system sufficient to protect the Indian Point Units from a land- or water-based attack. Request at 22. This request is improper for several reasons. First, it ignores that the Indian Point Units already has in place a security system that provides protection against land-based attacks, in compliance with NRC regulatory requirements. Second, it seeks that the Commission undertake actions that are not within its jurisdiction, such as instituting a no-fly zone around the Units (an action

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[14] Two recent decisions by NRC Atomic Safety and Licensing Boards have again confirmed that nuclear power plants need not make provisions to counter the malevolent use of airborne vehicles. Private Fuel Storage, L.L.C. (Independent Spent Fuel Storage Installation), LBP-01-37 (Memorandum and Order, December 13, 2001), slip op. at 12-14; Dominion Nuclear Connecticut, Inc. (Millstone Nuclear Power Station, Unit 3), LBP-02-05 (Memorandum and Order, February 5, 2002), slip op. at 13-18. The Commission has accepted referral of these and other recent licensing board decisions on terrorism, but its review is mainly focusing on whether the NRC has the obligation under the National Environmental Policy Act to consider intentional malevolent acts such as those directed against the United States on September 11, 2001 in its review of facility license applications. *See*, e.g., Private Fuel Facility, L.L.C. (Independent Spent Fuel Storage Installation), CLI-02-03 (Memorandum and Order, February 6, 2002), slip op. at 3.

[15] *See* note 4, *supra*.

that could only be taken by the Federal Aviation Administration [("FAA")]), and establishing a defense system in the plant's vicinity (an action that could only be implemented by the U.S. Air Force.) Third, Petitioners' request is outside the scope of §2.206, because establishing a no-fly zone and a defense system around the Indian Point site are not actions to "modify, suspend or revoke" the Indian Point Units' licenses. Fourth, it seeks modifications of Indian Point's security plan, *e.g.*, to institute anti-aircraft defense system, a request that is inappropriate for the reasons discussed in Subsections 1 and 2 above.

The NRC has limited jurisdiction and lacks statutory authority to unilaterally impose a no-fly zone around the Indian Point facility. Congress has given the Administrator of the FAA jurisdiction over the use of navigable airspace, including the establishment of flight paths and no-fly zones to protect property and national security.[16] 49 USC § 40103(b) (2001). The NRC does not have any such statutory authority. *See* 42 USC § 2201 (2001). At most, the NRC can use information received from the FAA when

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[16] 49 USC § 40103(b) provides:

(1) The Administrator of the Federal Aviation Administration shall develop plans and policy for the use of the navigable airspace and assign by regulation or order the use of the airspace necessary to ensure the safety of aircraft and the efficient use of airspace. The Administrator may modify or revoke an assignment when required in the public interest.
(2) The Administrator shall prescribe air traffic regulations on the flight of aircraft (including regulations on safe altitudes) for--
(A) navigating, protecting, and identifying aircraft;
(B) protecting individuals and property on the ground;
(C) using the navigable airspace efficiently; and
(D) preventing collision between aircraft, between aircraft and land or water vehicles, and between aircraft and airborne objects.
(3) To establish security provisions that will encourage and allow maximum use of the navigable airspace by civil aircraft consistent with national security, the Administrator, in consultation with the Secretary of Defense, shall--
(A) establish areas in the airspace the Administrator decides are necessary in the interest of national defense; and
(B) by regulation or order, restrict or prohibit flight of civil aircraft that the Administrator cannot identify, locate, and control with available facilities in those areas.

determining potential aircraft risks associated with the operation of a nuclear power plant. *See, e.g.,* Metropolitan Edison Co. (Three Mile Island Nuclear Station, Unit No. 2), ALAB-692, 16 NRC 921 (1982).[17] It is the FAA, however, the agency that regulates aircraft flight paths. *See, e.g.,* Consumers Power Co. (Big Rock Point Plant), DD-80-34, 12 NRC 711 (1980). Thus, the NRC could not grant the relief requested by Petitioner, even if such relief were warranted. By the same token, it is the U.S. Air Force, not the NRC, that has the statutory mandate to provide anti-aircraft protection measures.[18]

In sum, imposing and maintaining a no-fly zone around a nuclear power plant is not within the jurisdiction of the NRC, nor is an appropriate responsibility for the agency to undertake. Likewise, a licensee may not be charged with addressing such risks, for the reasons discussed in Subsections 1 and 2.

The same came be said with respect to Petitioners' request that the Indian Point site be protected against water-based attacks. It is the Coast Guard that has the responsibility to: "enforce U.S. laws on, under, and over the high seas and waters subject to the jurisdiction of the United States"; "engage in maritime air surveillance or interdiction to enforce or assist in the enforcement of the laws of the United States"; "administer laws and promulgate and enforce regulations for the promotion of safety of life and property on and under the high seas and waters subject to the jurisdiction of the United States"; and "maintain a state of readiness to function as a specialized service in

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[17] The NRC Standard Review Plan ("SRP"), NUREG-0800, calls for the performance at each nuclear power plant site of a probabilistic analysis of aircraft hazards to determine whether such hazards should be taken into account in the plant's design basis. NUREG-0800, §3.5.1.6. Such an analysis has been conducted for the Indian Point site, and is set forth in the FSARs for the Indian Point Units. The analyses conducted pursuant to the SRP guidance, however, address the probability of an accidental aircraft impact, not a deliberate act of terrorism.

[18] The Secretary of the Air Force has the responsibility for establishing and developing air defense installations and facilities that are necessary in the interest of national security. 50 USC § 491 (2001).

the Navy in time of war, including the fulfillment of Maritime Defense Zone command responsibilities." 14 USC § 2. There is no authority for the NRC to order the establishment of defenses against a potential water-based attack against the Indian Point Units, nor is there a need for such NRC action in light of the Coast Guard's explicit responsibilities.

4. The Request that the NRC Order the Revision of the Emergency Response Plans of Indian Point and Westchester County Goes Beyond the NRC Regulatory Authority and the Actions Sought Go Beyond those Required by the Regulations

Petitioners' fourth request for relief seeks to require Entergy and Westchester County to revise their emergency response plans to take into account and prepare for "possible terrorist attacks" on the Indian Point Units, including "comprehensive response to multiple attacks in the region which may impair efficient evacuation of the area." Request at 22-23. Again, this request asks the NRC to take actions beyond its authority and, to the extent that Petitioners seek to require Entergy to expand its existing emergency response plan beyond what is required by NRC regulations, amounts to an impermissible collateral attack on Commission regulations.

NRC regulations governing emergency response plans are set forth at 10 C.F.R. §50.47. *See* 10 C.F.R. §50.47 ("... no initial operating license for a nuclear power reactor will be issued unless a finding is made by the NRC that there is a reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency").[19] Petitioners' Request clearly goes beyond the requirements

[19] Further, NRC's review occurs in coordination with that of the Federal Emergency Management Agency ("FEMA"), which reviews a licensee's plan in conjunction with available municipal and state plans. *See* 10 C.F.R. §50.47(a)(2). These multiple levels of review provide assurance that suitable emergency response programs for a nuclear power plant and the surrounding communities are developed.

set forth in that regulation in two respects. First, it would require that Westchester County's Radiological Emergency Preparedness Plan (the "County Plan") be amended to address a number of contingencies, including terrorist attacks against the Indian Point Units, "Class 9 events", spent fuel storage releases, spent fuel assembly fires, and explosions at the Indian Point Units. Second, Petitioners would require Entergy to expand its emergency response plan to address not only terrorist attacks against Indian Point, but also multiple terrorist attacks at locations other than the Indian Point site, such as "destruction or blockage of the Tappan Zee bridge." Id.

As to the first issue, the Commission lacks authority to require Westchester County to make changes to the County Plan. Indeed, the provisions of 10 C.F.R. §50.47 apply only to licensees, such as Entergy, not to state or local governments. Moreover, 10 C.F.R. §50.47(c)(1)(iii) sets forth the Commission's well-settled "rule of realism," which recognizes that state and local government officials may not participate in emergency planning for a nuclear reactor. NRC regulations allow a licensee's emergency response plan to be deemed adequate despite this lack of participation, rather than requiring the state and local authorities to participate. *See, e.g.,* Public Service Company of New Hampshire (Seabrook Station, Units 1 and 2), LBP-89-32, 30 NRC 375 (1989). In so doing, the regulations implicitly recognize that the NRC lacks authority to direct the formulation, let alone the content, of emergency plans by local governments.

As to the second issue, Petitioners' request exceeds the requirements of NRC regulations by demanding that Entergy take into account possible acts of terrorism against other facilities in the region, and the manner in which such acts may affect the Indian Point emergency response plan. Again, the NRC regulations impose emergency response requirements on nuclear power plant licensees with respect to their facilities, and those alone. *See* 10 C.F.R. §50.47. Other agencies are responsible for developing

emergency plans for external facilities. See, e.g., NUREG-0645/FEMA-REP-1 (Rev. 1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants."[20]

The multiple terrorist attacks in the Indian Point region that Petitioners would have Entergy address are hostile acts against the United States. Therefore, under the Commission's long-standing policy discussed above (in Subsections 1 and 2), such attacks are expected to be addressed by the national security authorities and need not be part of the Units' design basis.

To the extent that Petitioners are seeking that Entergy confirm to the NRC that it meets current emergency response planning requirements, such an action is unnecessary. Emergency response plans for the four counties contiguous to the Indian Point site have been developed and approved by the Federal Emergency Management Agency in May 1996. The plans are regularly tested during drills and exercises. The licensee's Emergency Response Plans are periodically reviewed by the U.S. Nuclear Regulatory Commission and their compliance with the requirements of 10CFR Part 50.47 is verified.[21] Thus, Entergy is in compliance with NRC regulations, and there is no basis for the relief sought by Petitioners.

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[20] NUREG- 0654, Rev. 1, Supp. 1, provides that the offsite response organizations' plans to implement protective measures for the plume exposure pathway shall include, for example, the identification of and means for dealing with potential impediments (e.g., seasonal impassability of roads) to use of evacuation routes, and contingency measures. It also provides that the plant licensee is not responsible for providing emergency response measures (e.g., evacuation plans) for such contingencies where the state or local government has emergency response plans in place.

[21] *See, e.g.*, NRC Inspection Report No. 05000286/2000-10 (December 8, 2000).

5. The Request that the NRC Order Entergy to Convert the Current Spent Fuel Technology from a Spent Fuel Pool to a "Bunkered" Dry Cask System Seeks Actions not Required by the Regulations

Petitioners' fifth request for relief seeks an NRC order to convert the existing Indian Point spent fuel storage facilities to "bunkered" dry cask systems, on the grounds that the NRC has "never established that the Indian Point spent fuel storage facility is secure against foreseeable attacks," including a land, water and airborne "assault." Request at 24. This request is without basis in NRC regulations. Spent fuel pools are NRC-approved, "design basis" facilities, subject to continuous NRC regulation and oversight, and their protection against external threats is part of a nuclear facility's safeguards contingency plan. Further, there are currently no regulations requiring that dry storage systems be selected over spent fuel storage pools; indeed, Petitioners concede that no approved dry storage facilities exist with the design features called for in their Request, and no basis in the NRC regulations for requiring that such facilities be developed.[22]

In short, Petitioners' fifth request must be rejected as an effort to compel Entergy to take action beyond existing legal requirements.

6. The Request that the NRC Order the Retirement of the Indian Point Units is Unripe and Unwarranted

Petitioners' sixth request for relief asks the NRC to permanently shut down the Indian Point facility. Id. at 2. This request for relief is inappropriate and unwarranted,

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[22] The dry cask system proposed in the Request was apparently defined only in a telephone conversation between an unnamed representative of Petitioners and one Ed Lyman, described as a "Nuclear Physicist at Nuclear Control Institute." Request at 24. The system would consist of dry storage of the fuel in "robust steel casks that are cooled by natural circulation of air," with each cask "surrounded by an earth and gravel berm, with substantial spacing between the casks." Petitioners do not define the design requirements for such a system, other than indicating vaguely that the design basis for this storage arrangement "could include a requirement, among other things, that the impact of a fuel-Laden aircraft on the storage facility would not lead to a release of radioactive material from more than one cask." Id.

given Petitioners' lack of any legally cognizable basis for even seeking *suspension* of the Indian Point Units' respective operating licenses, as discussed in Subsection 1 above.

Moreover, even if any of the Petitioners' allegations were legitimate -- which they are not -- Petitioners' request should be denied as premature. Before permanent shutdown could result, Entergy would have to have an opportunity, pursuant to 10 C.F.R. §2.202, to address Petitioners' allegations, and to request a hearing, among other procedures. *See* 10 C.F.R. §2.202.Permanent shutdown of the Indian Point Units could only be achieved after the issues leading to the potential suspension of the Units' respective licenses had been addressed, the NRC had occasion to examine the actions taken by Entergy, and a determination could be made whether continued operation of the Units was consistent with public health and safety. None of those circumstances exist at the present time; thus, the request is not ripe for consideration. *See, e.g.*, General Public Utilities Nuclear Corporation (Oyster Creek Nuclear Generating Station), DD-97-14, 45 NRC 472 (1997) (a petitioner's request for 2.206 relief during transfer of spent fuel before licensee requested authorization to transfer the fuel was premature).

IV. THE RELIEF REQUESTED IS UNNECESSARY

A. THE COMMISSION IS ALREADY LOOKING AT TERRORISM ISSUES ON A GENERIC BASIS

As discussed earlier, the Commission has announced that it is conducting a "thorough review" of its safeguards and physical security programs, "from top to bottom." Private Fuel Storage L.L.C., *supra*, slip op. at 3-4. Out of that review, new requirements may emerge that would be applicable to all nuclear power plants, including the Indian Point Units. Any action that could be taken at this point with respect to Indian Point could be repetitive, or even inconsistent, with the results of the Commission's considered review of nuclear power plant safety issues. Therefore, any relief granted in

response to Petitioners' Request would be unnecessary, inappropriate, and potentially counterproductive.

B. THE ISSUES RAISED IN THE REQUEST ARE BEST ADDRESSED IN RULEMAKING

In the last several months, the NRC has repeatedly stated that it is performing a thorough, generic review of nuclear power plant security. See, e.g., Private Fuel Storage L.L.C., *supra,* CLI-01-26, slip op. at 3-4. It is well-established that generic safety questions should be resolved in a rulemaking rather than adjudicatory proceedings. See Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), CLI-74-40, 8 AEC 809, 814-15, clarified, CLI-74-43, 8 AEC 826 (1974). When an issue affects nuclear reactors generally, the proper approach is to petition the Commission to promulgate an amendment to its rules under 10 CFR § 2.802. See, e.g., Cleveland Electric Illuminating Co. (Perry Nuclear Power Plant, Units 1 and 2), LBP-81-57, 14 NRC 1037, 1038-1039 (1981); Metropolitan Edison Company (Three Mile Island Nuclear Station, Unit No. 1), 11 NRC 674, 675 (1980).

In this case, Petitioners could wait for the Commission to complete its review. If that review results in additional requirements, the Commission will likely initiate a proposed rulemaking on further nuclear power security measures. Petitioners could participate in such a proceeding as provided in 10 C.F.R. §2.805. Alternatively, they could file a rulemaking petition with the NRC pursuant to 10 C.F.R. §2.802 to seek institution of specified measures. Because these avenues for raising their concerns are available, Petitioners do not need relief under §2.206. Their Request is, therefore, unnecessary.

V. NO EVIDENCE HAS BEEN PRESENTED BY PETITIONERS THAT PLANT SHUTDOWN IS NECESSARY TO ADDRESS THE THREATS THEY POSTULATE

A. THERE IS NO IMMINENT THREAT TO THE INDIAN POINT UNITS THAT REQUIRES PLANT-SPECIFIC ACTIONS BEYOND ADDRESSING THE DESIGN BASIS THREAT

1. Alleged General Plant Vulnerability

a. Alleged Vulnerability of Control Room to Attack, Fire; Diesel Generator Vulnerability/Reliability

Petitioners allege that the Indian Point Units' respective control rooms "are a likely and vulnerable target for terrorist attack," and also that "[s]eizure or disability of the control rooms would dramatically increase the potential for the intentional or accidental destruction of the reactor core." Request at 6, 20. However, Petitioners provide no factual basis for their claim that there is a significant risk of terrorist seizure of the control room of either Indian Point Unit, or that such takeover would result in the destruction (either deliberate or accidental) of the reactor core.

As to the first claim regarding a control room seizure by terrorists, Petitioners fail to recognize that the control room of a nuclear power plant is defended by the same security measures that protect the entire plant. Thus, the control rooms are no more vulnerable than any other element of the Units to attack by terrorists. Also, each control room is contained within a building on site. Therefore, not only do the security barriers and the security force have to be overcome; the attackers must also successfully break into the building and seize the control room. Thus, a series of offensive actions would have to be accomplished successfully, by knowledgeable terrorists, before any damage related to the control rooms could occur.

Notwithstanding Petitioners' claims, the control rooms for the Indian Point Units are adequately protected against terrorist attacks. Unsupported allegations about their

vulnerability provide no basis for shutting down the Indian Point Units or instituting any of the other measures Petitioners seek.

Even assuming that the control room of one of the Indian Point Units were seized, it is very unlikely that Petitioners' predictions of damage to or destruction of the reactor would materialize. Nuclear reactors (including those at the Indian Point Units) must be designed so that if the controls are manipulated to an unsafe position or damaged, and reactor instrumentation detects that the reactor is entering a dangerous operating regime, the reactor will automatically shut down through the insertion of the reactor control rods into the core. (These requirements are specifically incorporated into the licensing bases of Indian Point Units 2 and 3, as set forth in the respective Final Safety Analysis Reports ("FSARs").) Core cooling systems will then ensure that the core is adequately cooled and maintained in a safe condition. The Indian Point Unit 2 and Unit 3 FSARs describe how the plants were designed to physically meet the aforementioned requirements.

Petitioners further claim—also without factual basis—that a terrorist attack might result in the failure of the control room functions due to a loss of off-site power, the failure of backup diesel generators, or the evacuation of the control room due to fire. Request at 6. None of these allegations is meritorious. Reactors are required to have redundant on-site and off-site power systems so that power to the reactor is maintained if one of the systems fails. Reactors also are required to be designed so that in the event of a "station blackout," in which the plant loses off-site and on-site power for a specified period of time, the reactor will automatically shut down and plant safety systems will maintain the core in a safe condition. *See* 10 C.F.R. §50.63(a) (loss of all alternating current power). The Indian Point Units satisfy these regulatory requirements. One of the Indian Point Units uses a gas turbine and the other, a separate diesel generator (apart from the emergency diesel generators) to provide electric power to critical plant facilities in the

unlikely event of a "station blackout". The FSARs for the Indian Point Units discuss how the plants address the "station blackout" rule.

It is unclear whether Petitioners' claim about diesel generators being "vulnerable and sometimes unreliable" (Request at 6) is directed at diesel generators in general or specifically at the diesel generators for the Indian Point Units. In either case, the claim is invalid. Diesel generators are safety systems subject to stringent NRC requirements, including those regarding quality control, maintenance, and testing. The Indian Point Units diesel generators have been demonstrated to meet all NRC regulatory requirements. Redundancies exist in the diesel generator configurations for the Indian Point Units so that, in the unlikely event their operation is needed but one failed, the Units would still have sufficient diesel generator capability. A variety of periodic tests, some weekly, some monthly, some quarterly, are performed on the diesel generators. These tests ensure that the diesel generators will be capable of operating when required.

To the extent that Petitioners are asserting that the Indian Point diesel generators are "vulnerable" in the sense of being subject to a terrorist attack, the diesel generators are provided the same degree of security protection as other plant safety-related structures, and as such are guarded against a terrorist attack. Also, the diesel generators are contained within a building onsite. Access to the diesel generator building is controlled and limited to specific personnel. Therefore, as with the control rooms, not only would the access barriers to the site and the security force have to be overcome, but entry into the building would have to be accomplished by knowledgeable terrorists before any damage could occur. In addition, Petitioners appear to be postulating the simultaneous loss through terrorist action of all off-site and on-site redundant sources of electric power, a threat without credibility or factual basis.

Finally, Petitioners' claim about the control rooms' vulnerability to fire is also unsupported and fails to recognize that NRC fire protection requirements specifically mandate that the reactor controls and the systems necessary to shut down the reactor must be protected from fire. 10 C.F.R. Part 50, App. R, Sec. III.G.1.a. Furthermore, control rooms must remain habitable so that reactor operators have time to shut down the reactor in the event of an onsite emergency, including a fire. *See* General Design Criterion 19; Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, AEC Regulatory Guide 1.78 (June 1974). The FSARs for the Indian Point Units describe the control room habitability features that exist to meet the applicable requirements and also describe how the Indian Point Units comply with fire protection requirements. The regulations further require that nuclear power plants have the capability to bring the reactor to a safe shutdown condition from outside the control room, so that in the event the control room has to be evacuated, safe shutdown of the plant can be achieved. The Indian Point Units meet these requirements. The FSARs for the Indian Point Units describe the Units' alternate shutdown capabilities from outside the control room.

As the preceding discussion demonstrates, the concerns by Petitioners about the potential damage to the Indian Point reactors from a terrorist attack that affects the control rooms are without basis and do not reflect any potential conditions that are not adequately addressed in the current plant design and operation.

b. Alleged Potential Breach of Containment Caused by Accidental or Intentional Crash of Fuel Laden Jetliners

Petitioners contend that the Indian Point reactors should be shut down because "the reactor containment walls were not designed to withstand the accidental or intentional crash of fuel laden jetliners." Request at 6. In particular, Petitioners rest their

claim on a 1982 Argonne National Laboratory ("ANL") study, which Petitioners claim showed that a crashing jetliner could breach the containment building of a nuclear plant and cause severe fire and explosion damage to the reactor. Id. at 12.

Contrary to Petitioners' allegations, the ANL study, Evaluation of Aircraft-Crash-Hazards Analyses for Nuclear Power Plants, NUREG/CR-2859 (June 1982) ("ANL Study"), does not show that a crashing jetliner could breach the containment building walls of reactors such as the Indian Point Units. The ANL study cites an estimate of the speed at which an airliner would have to impact the outer containment wall of a boiling water reactor ("BWR") in order to penetrate it.[23] The Indian Point units, however, are pressurized water reactors ("PWRs") with different containment design characteristics than those addressed in the ANL study. As described in the respective FSARs, the side walls of the containment cylinder and the dome are a minimum of 4-ft 6-in and 3-ft 6-in, thick, respectively, for Indian Point 2 and Indian Point 3.. Petitioners identify no evidence that a crashing aircraft would breach such thick walls. Therefore, the ANL study is not appropriately applicable to the Indian Point Units.

A crashing aircraft that breached the containment wall would be unlikely to cause damage to the reactor through an aircraft fuel fire.[24] In any event, the ANL study notes that fires would probably burn quickly, in minutes or tens of minutes, and the hazard from them would "appear to be tolerable in many instances." Id. at 75. While the study

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[23] While the ANL study addresses the consequences of aircraft crashes, it does not analyze them; rather, it summarizes and sometimes synthesizes other analyses that had been performed at the time of the study. See ANL Study at 61-78. Thus, in some cases, data important to assessing the applicability of the study to a real situation are missing. The thickness of the containment wall and the weight of the aircraft are examples of the missing data.

[24] The ANL Study discusses the potential for aircraft fuel explosions damaging the reactor only in the context of fuel vapor being trapped between the inner and outer containment of a BWR after the aircraft has penetrated the outer containment. ANL Study at 75-76. PWRs such as the Indian Point Units do not have double containment systems.

suggests examining crashes and the local impact areas for "unique situations which may cause an unacceptable hazard," id. (emphasis added), the remaining discussion of what would happen in the event of an aircraft crash is largely speculative, and is of doubtful applicability to Indian Point.[25]

In short, the ANL study does not support Petitioners' allegations that the Indian Point Units cannot sustain an aircraft crash. Therefore, neither the Study, nor any other materials referenced by Petitioners, support their position that the Indian Point Units should be shut down.

c. Alleged Need for Shutdown to Test Security of Spent Fuel

Petitioners call for "provisional" shutdown of the operating Indian Point Units to test "critical security provisions" with respect to their spent fuel pools ("SFPs"). Request at 8. This request, however, fails to consider the design and operating features of the SFPs.

The pools themselves are passive structures housing fuel elements in arrays of fixed racks. Pool cooling and makeup systems operate based on the amount of spent fuel in the pool and in accordance with established regulatory requirements, without regard for the operating status of the associated reactors. Therefore, no change in pool operations would result from a shutdown of the Units. There is, likewise, no change in either the type or extent of security protection for the SFPs of operating reactors such as the Units based on the reactors' operating status.

The spent fuel pool facility's operational and security requirements, therefore, are independent of the operating status of the reactor units. Petitioners' implications

[25] See id. at 76-78 (noting, e.g., at 76, that "the dissemination of the fuel and its partial mixing with the surrounding air to form an explosive cloud are virtually impossible to predict with any acceptable degree of accuracy").

otherwise are unfounded. Thus, no basis exists for a plant shutdown to "test" spent fuel pool security, since there is nothing to test and, even if something needed testing, such a test would not require a plant shutdown.

2. Alleged Vulnerability of Spent Fuel Pool

Petitioners' request for the shutdown of the Indian Point Units rests, in large measure, on the unsupported factual assertion that "[t]errorist action against the spent fuel storage facility could result in a catastrophic failure of the containment system." Request at 8. This argument, however, rests on a mischaracterization and improper use of an NRC Staff Technical Study and on incomplete understanding of the physical characteristics and features of the SFPs for the Units.

a. The NRC Staff's Technical Study of Spent Fuel Accident Risks Does Not Support Petitioners' Allegations Regarding SFP Vulnerability

Apparently the sole source of Petitioners' assertions with regard to SFP vulnerability to terrorist attacks is an October 2000 NRC Staff Technical Study of SFP accidents.[26] Petitioners, however, overlook that the Technical Study evaluates the risks from SFP operations at *plants undergoing decommissioning* to "identify the design and operational features necessary to ensure that the risks to the public from these *shutdown facilities* are sufficiently small."[27] Thus, the analyses in the Technical Study are not applicable to the Indian Point facility because, as the study makes clear, "as-operated SFP cooling systems [at decommissioning facilities] were different from those in operation

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[26] U.S. NRC, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants" (Oct. 2000) ("Technical Study").

[27] Id. at 1-1 (emphasis added).

when the plants were in power operation."[28] The Technical Study points out that, at a facility undergoing decommissioning,

> [t]he operating plant pool cooling and makeup systems generally have been removed and replaced with portable, skid-mounted pumps and heat exchangers. In some cases there are redundant pumps. In most cases, physical separation, barrier protection, and emergency onsite power sources are no longer maintained.[29]

Such conditions, which may reduce a licensee's ability to respond to an event resulting in damage to an SFP, are not allowed at operating plants and do not exist at Indian Point Units 2 and 3. Thus, Petitioners' reliance on the Technical Study is misplaced from the start.

In contrast to the limited SFP support systems assumed in the Technical Study to be available at a decommissioned plant, the Indian Point SFP support systems are fully operational and compliant with Commission safety regulations. The FSARs for Indian Point 2 and Indian Point 3 provide a description of the respective Fuel Handling Systems, including the SFPs, and their support systems.

b. Vulnerability of the Indian Point SFP to Aircraft-Induced Zirconium Fires

Petitioners allege that "a likely result" of an aircraft crashing into a SFP, or a truck bomb explosion, "would be a precipitous loss of cooling water in the spent fuel pools." Request at 9.[30] No support is given for this conclusory statement, which is contradicted in the Request itself by the assertion that the potential consequences of an attack against a SFP have not been evaluated. Id. at 8. Instead of offering any analysis,

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28 Id. at 3-2.

29 Id.

30 Truck bomb explosions are discussed separately below.

Petitioners transform a Technical Study definition of the term "catastrophic damage" (i.e., what would have to occur for damage to a SFP to be considered catastrophic) into an allegation as to what would happen if an aircraft crash occurred.[31] Id. at 11 ("probability is based on the occurrence of catastrophic damage to the spent fuel pool"). Petitioners also reference the Technical Study to support their assertion that the Indian Point "spent fuel storage area is highly vulnerable to an air attack and mitigation and control of damage from such an attack is highly improbable." Id. at 10. However, the Technical Study does not contain any basis for these conclusions, which are unfounded.

Petitioners speculate that the inevitable effect of a terrorist attack using an aircraft against a spent fuel pool would be an uncontrolled zirconium exothermic reaction (i.e., a zirconium "fire"). Id. at 9. However, a zirconium "fire," if possible at all, requires specific conditions highly unlikely to result from an aircraft crash into the spent fuel pool. The Technical Study identifies that the onset of a zirconium "fire" is dependent on heat generation and losses.[32] Normally, water in the spent fuel pool maintains the fuel rods well below the temperature for the postulated zirconium reaction to occur. Only if the water is lost from the spent fuel pool and the fuel remains uncovered for an extended time is it possible for the zirconium cladding of some spent fuel elements to heat up sufficiently to initiate an exothermic reaction. The Technical Study states that, assuming a *complete* and *instantaneous* draining of a spent fuel pool, approximately 10 hours is required for a spent fuel element to heat up to the minimum reaction temperature.[33] If the postulated aircraft crash does not drain the pool below the top of the fuel, a zirconium

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31 Catastrophic damage means such damage that "the pool ... rapidly drains and cannot be refilled from either onsite or offsite resources."

32 Technical Study at A1B-1.

33 Id.

"fire" is not possible at all. Without the thermal plume from a large self-sustaining zirconium fire, the fission products in the spent fuel are not dispersed offsite and would only be of onsite concern.[34] The sequence of unlikely events required to cause a zirconium fire that leads to offsite radiation releases is, at best, a highly improbable result of a highly improbable aircraft crash.

Indeed, it is difficult to conceive of the manner in which an aircraft crash could produce the specific conditions required for an exothermic reaction to occur in a SFP. As described in the Technical Study, spent fuel pools at nuclear power plants are structures constructed of thick, reinforced concrete walls. (As set forth in the FSARs for the Indian Point Units, the concrete walls of the Indian Point Units' spent fuel pools are 3 to 6 feet thick, and the pools are lined with stainless steel liners .25 inches thick.) The pools are approximately 40 feet deep, are partially embedded in the ground, and are designed to withstand severe earthquakes. Because of these and other design features, the Indian Point spent fuel structures, like similar "structures at [other] nuclear power plants, are able to withstand loads substantially beyond those for which they were designed."[35]

Any aircraft impact must breach these robust structures in order to cause a loss of pool water. An aircraft would have to penetrate all the way through the interior steel liner, not just crack the concrete, to cause a leak. The breach of either barrier is unlikely; both breaches occurring as a result of a single event is highly unlikely. Moreover, even if a breach of the pool did occur, the leak rate would have to exceed the capacity of all of the redundant makeup water systems in order to drain the pool.

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34 Id. at 3-1.

35 Technical Study at 3-19.

Petitioners also ignore the Technical Study's very conservative *assumption* that the SFP is a point target that sustains a "direct hit." These assumptions are crucial because, if "instead of a direct hit, the aircraft skids into the pool or a wing clips the pool, catastrophic damage may not occur."[36] The Indian Point SFPs are housed in the fuel handling buildings, which are relatively small structures shadowed by the respective containment buildings, which are located right next to them. Other buildings near the fuel handling buildings also provide sheltering, making the flight path for a direct hit on the fuel pool, if possible at all, a difficult feat for even the most experienced pilot.

The Technical Study also analyzed the thermal-hydraulic characteristics of spent fuel stored in SFPs to "determine the time available for plant operators to take actions to prevent a zirconium fire" and separated that time into two periods: the time available before fuel uncovering and that available before the zirconium might ignite.[37] The Staff determined that, assuming the loss of all SFP cooling and makeup systems, the minimum time to heat up and pool draining through water boil off to within 3 feet of the top of the spent fuel was 100 hours (more than 4 days). Further, the analysis methods used by the Staff were such that "the time available for fuel handler recovery from SFP events before initiation of a zirconium fire is underestimated."[38] At least another 10 hours must elapse before any zirconium ignition can occur. If the spent fuel stored in a pool is aged more than 5 years, zirconium ignition is impossible unless air flow to the fuel elements is obstructed and accident management measures are unsuccessful.[39]

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36 Id. at A1B-1.

37 Id. at 2-1.

38 Id. at 3-1 - 3-2.

39 Id. at 2-2.

The Technical Study also identified that the risk from loss of spent fuel pool water was "quite sensitive to the performance of the SFP operating staff in identifying and responding to" the event.[40] It is unreasonable to assume that the plant staff would fail to recognize and respond to an aircraft crash in the SFP area. The Technical Study modeled a SFP area fire including the conservative assumptions that SFP cooling systems will be irreparably damaged within 20 minutes and any electrically-driven fire pump is unavailable.[41] Despite these conservative assumptions, the Staff calculated the conditional "fuel uncovery frequency" for a SFP area fire as 2.3×10^{-5} per year.[42] In other words, for every 43,000 aircraft crashes into the SFP at a facility undergoing decommissioning, one *might* result in the uncovering of the spent fuel and a potential zirconium fire, even assuming a host of extreme conditions to be present. At operating plants with redundant SFP cooling systems, such as the Indian Point Units, fuel uncovering is even less likely.

In short, Petitioners' claim that a reduction in SFP water level "will lead to a spent fuel rod assembly fire" (Request at 9) is erroneous and unsupported by the Technical Study. To the contrary, the Technical Study – while germane only to units undergoing decommissioning – concluded that even at facilities with reduced SFP support systems, the "results of the study indicate that the risk at SFPs is low and well within the Commission's Quantitative Health Objectives."[43] In fact, the Technical Study reached quite the opposite conclusion to that alleged by Petitioners:

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40 Id. at 3-5.

41 Petitioners erroneously point to these conservative and restrictive assumptions as the technical bases for their assertion that a zirconium fire in one pool would "quickly cause fires in other pools where water loss is occurring." Request at 10.

42 Technical Study at 3-16, A23a-31.

43 Technical Study at viii.

> The risk at decommissioning plants is low and well within the Commission's safety goals. The risk is low because of the *very low likelihood of a zirconium fire* even though the consequences from a zirconium fire could be serious.[44]

For these reasons, the threat to the SFP at the Indian Point Units from an aircraft crash into their spent fuel pools is insignificant, and the Petitioners' allegations are without factual basis.

c. A Potential Truck Bomb Attack against the SFP is Addressed as Part of the Design Basis Threat

Existing Commission regulations explicitly describe protective measures licensees must implement against "truck bombs." A four-wheel drive land vehicle bomb is specifically identified in 10 C.F.R. §73.1(a)(1)(E)(iii) as part of the design basis threat. The implementing rules require "vehicle barrier systems must be established to protect against use of a land vehicle" as a "means of transportation to gain unauthorized proximity to vital areas." 10 C.F.R. §73.55(c)(7). These barriers are erected so that a truck can not penetrate to a point where an explosion could damage the reactor or other critical reactor safety systems, including SFPs. In addition, the alarmed fence would have to be penetrated by the intruders. Both the vehicle barrier system and the alarmed fence should slow down any truck long enough for the security force to take appropriate action.

Even if a "truck bomb" could somehow be placed near the Indian Point SFP, the same design and mitigation features discussed above with respect to aircraft crashes make it highly unlikely that such an explosive device could cause sufficient damage to drain the pool, and even more unlikely that a zirconium fire would result.

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[44] Id. at x, 5-3 (emphasis added).

In summary, the threat to the SFP at Indian Point from an aircraft crash or truck bomb on the spent fuel pool is insignificant and the Petitioners' allegations are without factual basis. At most, Petitioners may have raised a possible generic issue, which if found appropriate for study, should be addressed on an industry-wide basis. A shutdown of the Indian Point Units to evaluate the issue is unnecessary and inappropriate.

B. ENTERGY, THE STATE AND THE LOCAL AUTHORITIES ARE CAPABLE OF RESPONDING TO A TERRORIST THREAT, AS DEFINED BY REGULATIONS

Petitioners launch unsupported, and largely irrelevant, attacks against Entergy's ability to provide effective security for the Indian Point Site.[45] However, as described above, the Indian Point Units have security plans that meet regulatory requirements and are capable of responding to a design basis terrorist threat. Likewise, the emergency response plans of Entergy and the counties adjacent to the Indian Point site comply with applicable requirements and have been reviewed and approved by the NRC and FEMA, respectively. Thus, there can be no doubt that the organizations in charge of security and emergency planning activities relating to Indian Point are fully capable of discharging their responsibilities.

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[45] Thus, Petitioners argue that "[a]s recently as August 2000, Entergy was sanctioned by the NRC for failure to maintain adequate physical protection of the Waterford 3 facility in Killona, Louisiana." Request at 14. Since Petitioners do not allege that the same personnel (or even the same management team) involved in the alleged violations at Waterford is also responsible for security at the Indian Point Units, these accusations, even if correct, would be irrelevant to the adequacy of security provisions at another plant a thousand miles away.

Likewise, Petitioners allege that "[l]ast year, Indian Point 2 became the first nuclear plant in the nation to be given a "red" designation, giving it the highest risk assessment in the nation. The NRC gave the plant its worst rating because of the operators failure to detect flaws in a steam generator tube before a radiation leak in February 2000." Id. Yet, at the time the incidents alleged by Petitioners occurred, the Indian Point Units were not being operated by Entergy, and at any rate failure to detect steam generator tube flaws is not a plant security issue. Therefore, these claims are also irrelevant on their face.

VI. CONCLUSION

In their Request, Petitioners call for the immediate shutdown of the Indian Point Units, the immediate transfer of the fuel in the spent fuel pools into dry cask storage, and drastic expansions of Indian Point's security provisions and the emergency response plans for the site. This extraordinary relief is unjustified.

Petitioners seek this relief, not on the basis of any non-compliance by Entergy with existing law or NRC regulations, but on the theory that Indian Point could be the subject of terrorist attacks such as those conducted on September 11, 2001 by foreign enemies. Petitioners ignore, however, the long-standing Commission policy to rely on our national security apparatus to defend nuclear facilities against attacks by enemies of the United States. The Commission has done so because that is the will of Congress, because it is not practical to turn nuclear facilities into armed fortresses, and because the Commission and nuclear facility licensees should not be required to speculate about enemy threats and the effectiveness of the national security apparatus in defending against them.

The Request ignores the increased security measures that have been implemented since September 11 at our airports, seaports, borders, and coastlines. It ignores the war in Afghanistan and other worldwide efforts against terrorism, and the ongoing law enforcement actions against suspected terrorists here. It also gives no credit to the additional security measures that have been implemented at Indian Point since the attacks. In so doing, the Request improperly exaggerates the extent of the terrorist threat and downplays the ability to defend against that threat.

Furthermore, Petitioners ignore that in light of the September 11 attacks, the Commission is conducting a widely publicized, top-to-bottom review of its security requirements. Because security is a generic issue best addressed through rulemaking, it

would be wholly inappropriate to entertain a 2.206 petition on the same subject addressed to a single facility.

Finally, the Petition postulates a series of terrorist attack scenarios leading to hypothetical releases of radioactive materials to the public. Those release scenarios are alarmist, based on erroneous factual assumptions and misinterpreted references, and so improbable as to lack credibility.

In short, Petitioners have asserted no legal grounds and produced no factual information that would warrant emergency action at Indian Point. Their Request, therefore, must be denied.

Respectfully submitted,

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Dated: February 11, 2002

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Director, Office of Nuclear Reactor Regulation

In the Matter of	)	
	)	
ENTERGY NUCLEAR INDIAN POINT 2, LLC,	)	Docket Nos. 50-247
ENTERGY NUCLEAR INDIAN POINT 3 LLC,	)	and 50-286
and ENTERGY NUCLEAR OPERATIONS, INC.	)	(License Nos. DPR-26
	)	and DPR-64)
	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of the Response to Riverkeeper, Inc. 2.206 Request re the Potential Shutdown of Indian Point 2 and 3's were served on the persons listed on the attached Service List (unless otherwise noted) by U.S. mail, first class, postage prepaid this 11th day of February, 2002.

Jay E. Silberg

Jay E. Silberg

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