NUREG-1560
Vol. 3

Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance

Part 6

Appendices A, B, and C

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PART 6

Appendices

ABSTRACT

This report provides perspectives gained by reviewing 75 Individual Plant Examination (IPE) submittals pertaining to 108 nuclear power plant units. IPEs are probabilistic analyses that estimate the core damage frequency (CDF) and containment performance for accidents initiated by internal events (including internal flooding, but excluding internal fire).

The U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research, reviewed the IPE submittals with the objective of gaining perspectives in three major areas: (1) improvements made to individual plants as a result of their IPEs and the collective results of the IPE program, (2) plant-specific design and operational features and modeling assumptions that significantly affect the estimates of CDF and containment performance, and (3) strengths and weaknesses of the models and methods used in the IPEs. These perspectives are gained by assessing the core damage and containment performance results, including overall CDF, accident sequences, dominant contributions to component failure and human error, and containment failure modes. In particular, these results are assessed in relation to the design and operational characteristics of the various reactor and containment types, and by comparing the IPEs to probabilistic risk assessment characteristics. Methods, data, boundary conditions, and assumptions used in the IPEs are considered in understanding the differences and similarities observed among the various types of plants.

This report is divided into six parts. Part 1 is a summary report of the key perspectives gained in each of the areas identified above, with a discussion of the NRC's overall conclusions and observations (Chapter 8). Parts 2 through 6 provide a more in-depth discussion of the perspectives summarized in Part 1. Specifically, Part 2 discusses key perspectives regarding the impact of the IPE Program on reactor safety (summarized in Part 1, Chapter 2). Part 3 discusses perspectives gained from the IPE results regarding CDF, containment performance, and human actions (summarized in Part 1, Chapters 3, 4, and 5, respectively). Part 4 discusses perspectives regarding the IPE models and methods (summarized in Part 1, Chapter 6). Part 5 discusses additional IPE perspectives (summarized in Part 1, Chapter 7). Part 6 contains Appendices A, B and C whic provide the references of the information from the IPEs, updated PRA results, and public comments on draft NUREG-1560 (including staff responses), respectively.

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ABBREVIATIONS

BWR	Boiling Water Reactor
CCFP	Conditional Containment Failure Probability
CDF	Core Damage Frequency
Cs	Cesium
ECCS	Emergency Core Cooling System
HEP	Human Error Probability
HRA	Human Reliability Analysis
I	Iodine
IPE	Individual Plant Examination
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
MAAP	Modular Accident Analysis Program
NRC	Nuclear Regulator Commission
NSSS	Nuclear Steam Supply System
PECO	Philadelphia Electric Company
PRA	Probabilistic Risk Assessment
PSF	Performance Shaping Factor
PWR	Pressurized Water Reactor
QHO	Quantitative Health Objective
RCP	Reactor Coolant Pump
SER	Staff Evaluation Report
SRV	Safety Relief Valve
SBO	Station Blackout
Te	Tellurium
VP	Vice President
WOG	Westinghouse Owner's Group

APPENDIX A

INDIVIDUAL PLANT EXAMINATION REFERENCES

In this Appendix, the references for the Individual Plant Examination (IPE) submittals are provided. IPE submittals and responses to NRC request(s) for additional information are listed in Table A-1. This information is provided via the submittal's date and submittal's public document room accession number.

Table A-1 IPE References (of Information Used in NUREG-1560).

Plant name	Individual plant examination submittal		Licensee responses to request for additional information	
	Submittal Date	Accession Number	Submittal Date	Accession Number
Arkansas Nuclear One, 1	4/29/1993	9305040339	*	-
Arkansas Nuclear One, 2	8/28/1992	9209010212	10/5/1995	9510110067
Beaver Valley 1	10/1/1992	9210150272	3/10/1995	9503130375
Beaver Valley 2	3/17/1992	9203240301	9/11/1992 10/26/1992	9210010222 9211030334
Big Rock Point	3/27/1994	9406080120	*	-
Braidwood 1&2	6/30/1994	9408110123	*	-
Browns Ferry 2	9/01/1992 4/14/1995	9209030199 9504180280	9/21/1993 12/23/1993	9309280175 9401060224
Brunswick 1&2	8/31/1992	9209100204	9/09/1994 2/27/1995	9409200201 9503070179
Byron 1&2	4/28/1994	9405250189	*	-
Callaway	9/29/1992	9210090033	11/22/1995	9511280238
Calvert Cliffs 1&2	12/30/1993	9401070022	9/12/1995	9509150108
Catawba 1&2	9/10/1992	9209240287	6/07/1993	9306150372
Clinton	9/23/1992	9210050174	11/22/1995	9511300286
Comanche Peak 1&2	10/30/1992	9211050102	*	-
Cooper	3/31/1993	9304060035	2/20/1995	9502280017
Crystal River 3	3/09/1993	9303150193	11/22/1995	9511280382
Davis-Besse	2/26/1993	9303030295	9/11/1995	9509150145
DC Cook 1&2	5/01/1992 2/26/1993	9205050329 9303030121	2/24/1993 12/03/1993 4/25/1994	9303010355 9312030217 9405090139

Table A-1 IPE References (of Information Used in NUREG-1560).

Plant name	Individual plant examination submittal		Licensee responses to request for additional information	
	Submittal Date	Accession Number	Submittal Date	Accession Number
Diablo Canyon 1&2	4/14/1992	9204240011	1/15/1993	9301250130
Dresden 2&3	1/28/1993	9304130182	10/28/1994	9411010060
Duane Arnold	11/30/1992	9212090167	6/26/1995	9507100196
Farley 1&2	6/14/1993	9306250041	11/09/1994	9411180035
Fermi 2	9/01/1992	9209090121	6/30/1994	9407060029
FitzPatrick	9/13/1991	9109190203	9/01/1992	9209140256
Fort Calhoun 1	12/01/1993	9312070021	11/30/1995	9512040426
Ginna	3/15/1994	9403230240	*	-
Grand Gulf 1	12/23/1992	9212290071	**	-
Haddam Neck	6/29/1993	9307070183	*	-
Hatch 1&2	12/11/1992	9212230136	10/07/1994	9410120348
Hope Creek	5/31/1994	9406060125	11/06/1995	9511090179
Indian Point 2	8/12/1992	9208200238	10/31/1995	9511210368
Indian Point 3	6/30/1994	9407120222	6/20/1995	9506290190
Kewaunee	12/01/1992	9212090115	1/13/1995	9501200288
LaSalle 1&2	4/28/1994	9405090227	**	-
Limerick 1&2	7/30/1992	9208030288	**	-
McGuire 1&2	11/04/1991	9111070233	6/30/1992 10/5/1992	9207080050 9210210155
Maine Yankee	8/28/1992	9208030288	2/28/1995	9503080175
Millstone 1	3/31/1992	9204070238	5/25/1993	9306030323
Millstone 2	12/31/1993	9401100239	5/31/1994 9/27/1995	9406070213 9509250347
Millstone 3	8/31/1990	9009100231	4/22/1991	9104290183
Monticello	2/27/1992	9203090231	2/15/1993	9302220084

Table A-1 IPE References (of Information Used in NUREG-1560).

Plant name	Individual plant examination submittal		Licensee responses to request for additional information	
	Submittal Date	Accession Number	Submittal Date	Accession Number
Nine Mile Point 1	7/27/1993	9308030002	6/26/1995	9507030056
Nine Mile Point 2	7/30/1992	9208050183	5/06/1993	9305130111
North Anna 1&2	12/14/1992	9212210199	4/27/1995	95050200379
Oconee 1,2&3	11/30/1990	9012060005	8/14/1992	9208240190
Oyster Creek	8/24/1992	9208280377	7/02/1993	9307150084
Palisades	1/29/1993	9302120094	7/22/1994	9407280168
Palo Verde 1,2&3	4/28/1992	9205060025	2/25/1993	9303020319
Peach Bottom 2&3	8/26/1992	9209010209	**	-
Perry 1	7/15/1992	9207240153	11/24/1993	9312060116
Pilgrim 1	9/30/1992	9210190105	12/28/1995	9601020192
Point Beach 1&2	6/30/1993	9307020355	9/26/1994	9406030077
Prairie Island 1&2	3/01/1994	9403090295	2/27/1996	9603040214
Quad Cities 1&2	12/13/1993	9312210240	8/08/1994 12/23/1994	9408120259 9412290313
River Bend	12/01/1993	9302120067	9/22/1995	9509260374
Robinson 2	8/31/1992	9209090152	9/27/1993	9309140049
Salem 1&2	7/30/1993	9308060186	*	-
San Onofre 2&3	4/29/1993	9305040246	1/20/1995	9501260308
Seabrook	3/01/1991	9103060219	7/23/1991	9107310374
Sequoyah 1&2	9/01/1992	9209030210	2/25/1994	9403080390
Shearon Harris 1	8/20/1993	9309010155	1/25/1995 9/18/1995	9501250408 9509250025
South Texas 1&2	8/28/1992	9209110105	11/17/94	9411300102
St. Lucie 1&2	12/09/1993	9312150124	*	-
Summer	6/18/1993	9306290220	3/20/1996	9603250275

Table A-1 IPE References (of Information Used in NUREG-1560).

Plant name	Individual plant examination submittal		Licensee responses to request for additional information	
	Submittal Date	Accession Number	Submittal Date	Accession Number
Surry 1&2	11/26/1991	9112060076	5/15/1992	9206010089
Susquehanna 1&2	12/13/1991	9112200133	6/27/1992	9202030122
Three Mile Island 1	5/20/1993	9305280148	12/6/1995	9512110427
Turkey Point 3&4	6/25/1991	9106280106	3/11/1992	9203170219
Vermont Yankee	12/21/1993	9401060043	*	-
Vogtle 1&2	12/23/1992	9212280069	9/13/1995 10/02/1995	9509190310 9510060072
Waterford 3	8/28/1992	9209010231	*	-
Watts Bar 1	9/01/1992 5/02/1994	9209030222 9405090112	12/27/93	9401070397
WNP-2	8/28/1992	9209080185	10/20/1995	9510230409
Wolf Creek	9/28/1992	9210050289	8/30/1995	9509060171
Zion 1&2	4/24/1992 9/01/1995	9204290315 9509080045	2/22/1993	9302250285

* Information not provided on time for this report
** Information not requested

APPENDIX B

INDIVIDUAL PLANT EXAMINATION UPDATES

The perspectives provided in this report are based on the original probabilistic risk analyses (PRAs) performed by the licensees for their Individual Plant Examinations (IPEs). In many cases licensees updated these analyses to reflect plant changes and, in some cases, to incorporate staff concerns, as noted in the staff evaluation report (SER) of the licensee's IPE. For some of these PRAs, the results (e.g., core damage frequencies (CDFs) and dominant sequences) changed. Furthermore, several licensees provided as part of their comments on Draft NUREG-1560 information regarding revised analyses and plant changes. These changes are not reflected in the body of this report; they are provided, however, in this Appendix.

Table B.1 summarizes the updated plant-specific information. Plant names are listed in the first column of the table; the CDF of the original IPE submittal is listed in the second column for those plants that an updated CDF was reported; the updated CDF is listed in the third column. Information regarding updated analyses or plant changes is summarized in the fourth column; and corresponding references are provided in the fifth column.

It is noted that if a licensee has reported an updated CDF more than once the most recently reported CDF is listed.

Table B-1 Updated plant-specific information[*]

Plant name	IPE CDF	Updated CDF	Comments	Reference
Arkansas Nuclear One, 1	**	**	**	-
Arkansas Nuclear One, 2	**	**	**	-
Beaver Valley 1	**	**	**	-
Beaver Valley 2	2.1E-4/yr	3.1E-5/yr	No additional information was provided	Workshop Presentation by Westinghouse Owners Group (WOG), April 8, 1997.
Braidwood 1&2	2.7E-4/yr	1.1E-5/yr	Update included: - modeling and data changes - complete revision of the human reliability analysis (HRA) - credit for several hardware and procedural improvement New dominant sequences were reported	Commonwealth Edison, "Byron & Braidwood Stations Individual Plant Examinations, Response to NRC Requests for Additional Information and Modified Byron and Braidwood IPEs," March 27, 1997. Commonwealth Edison Company, "Commonwealth Edison Company Comments Regarding Draft NUREG-1560," February 14, 1997.
Big Rock Point	**	**	**	-
Browns Ferry 2	**	**	**	-
Brunswick 1&2	2.7E-5/yr	9.2E-6/yr	It is stated that the CDF change is the result of modeling changes and plant improvements	Carolina Power & Light Company, "Comments on Draft NUREG-1560, Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance (61 FR 65248)," March 14, 1997.

Table B-1 Updated plant-specific information*

Plant name	IPE CDF	Updated CDF	Comments	Reference
Crystal River 3	**	**	The licensee provided an update of its IPE response to address the weaknesses noted in the SER	Florida Power Corporation, "Individual Plant Examination - Internal Events," July 11, 1997.
Davis-Besse	**	**	**	-
DC Cook 1&2	6.3E-5/yr	7.1E-5/yr	No additional information was provided	Workshop Presentation by WOG, April 8, 1997.
Diablo Canyon 1&2	8.8E-5/yr	4.5E-5/yr	No additional information was provided A sixth diesel generator was installed since the submittal	Workshop Presentation by WOG, April 8, 1997. Pacific Gas and Electric Company, "Response to Request for Comments on Draft NUREG-1560," March 10, 1997.
Dresden 2 Dresden 3	1.9E-5/yr 1.9E-5/yr	3.4E-6/yr 5.0E-6/yr	Update included: - modeling and data changes - complete revision of HRA - several hardware and procedural improvements New dominant sequences were reported	Commonwealth Edison Company, "Dresden Individual Plant Examination (IPE), Response to NRC Staff Evaluation Report and Modified Dresden IPE," June 28, 1996. Commonwealth Edison Company, "Commonwealth Edison Company Comments Regarding Draft NUREG-1560," February 14, 1997.
Duane Arnold	7.8E-6/yr	1.5E-5/yr	New dominant sequences were reported	IES Utilities, Inc., "Duane Arnold Energy Center, Response to Request for Additional Information on Individual Plant Examination," June 26, 1995.
Farley 1&2	1.3E-4/yr	9.2E-5	No additional information was provided	Workshop Presentation by WOG, April 8, 1997.
Fermi 2	**	**	**	-

Table B-1 Updated plant-specific information*

Plant name	IPE CDF	Updated CDF	Comments	Reference
LaSalle 1&2	4E-5/yr	1E-5/yr	The IPE has been updated and includes: - modeling and data changes - revision of the HRA - hardware improvement No additional information was provided	Commonwealth Edison Company, "Commonwealth Edison Company Comments Regarding Draft NUREG-1560," February 14, 1997.
Limerick 1&2	**	**	All improvements listed as planned in the IPE submittal have been implemented	PECO Energy Company, "Comments Concerning Draft NUREG_1560, Individual Plant Examination Program: Perspectives on Reactor Safety Performance," March 14, 1997.
Maine Yankee	**	**	**	-
McGuire 1&2	**	**	Direct current power improvements have been implemented and the PRA is being updated	Duke Power, "Duke Power Company Comments on Draft NUREG-1560," March 3, 1997.
Millstone 1	**	**	**	-
Millstone 2	**	**	**	-
Millstone 3	5.6E-5/yr	5.8E-5/yr	No additional information was provided	Workshop Presentation by WOG, April 8, 1997.
Monticello	**	**	**	-
Nine Mile Point 1	**	**	**	-
Nine Mile Point 2	**	**	**	-
North Anna 1&2	7.2E-5/yr	5.6E-5/yr	No additional information was provided	Workshop Presentation by WOG, April 8, 1997.

Table B-1 Updated plant-specific information*

Plant name	IPE CDF	Updated CDF	Comments	Reference
Oconee 1,2&3	**	**	The PRA has been updated	Duke Power, "Duke Power Company Comments on Draft NUREG-1560," March 3, 1997.
Oyster Creek	**	**	**	-
Palisades	**	**	**	-
Palo Verde 1,2&3	**	**	**	-
Peach Bottom 2&3	**	**	**	-
Perry 1	**	**	The contribution of anticipated transient without scram (ATWS) to the overall CDF has been reduced as a result of procedural modifications The two improvements inhibit automatic depressurization during an ATWS and passive containment vent, under consideration during the IPE, will not be implemented	Centerior Energy, "Perry Nuclear Plant Voluntary Comment on Draft 1560, Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance, Summary Report," February 25, 1997.
Pilgrim 1	5.8E-5/yr	2.8E-5/yr	New dominant sequences were reported	Boston Edison, "Response to Request or Additional Information Regarding the Pilgrim Individual Plant Examination (IPE) Submittal," December 28, 1995.
Point Beach 1&2	**	**	**	-
Prairie Island 1&2	5.0E-5/yr	1.7E-5/yr	No additional information was provided	Workshop Presentation by WOG, April 8, 1997.

Table B-1 Updated plant-specific information*

Plant name	IPE CDF	Updated CDF	Comments	Reference
Summer	2.0E-4/yr	9.6E-4/yr	No additional information was provided	Workshop Presentation by WOG, April 8, 1997.
Surry 1&2	1.25E-4/yr	7.2E-5/yr	No additional information was provided	Workshop Presentation by WOG, April 8, 1997.
Susquehanna 1&2	**	**	**	-
Three Mile Island 1	**	**	**	-
Turkey Point 3&4	**	**	**	-
Vermont Yankee	**	**	**	-
Vogtle 1&2	4.9E-5/yr	4.4E-5/yr	No additional information was provided	Workshop Presentation by WOG, April 8, 1997.
Waterford 3	**	**	**	-
Watts Bar 1	3.3E-4/yr	4.4E-5/yr	No additional information was provided	Workshop Presentation by WOG, April 8, 1997.
WNP-2	2.0E-5/yr	1.5E-5/yr	New dominant sequences were reported	Washington Power Supply System, "Response to Request for Additional Information Related to Washington Public Power Supply System (WPPS) Nuclear Project No. 2 (WNP-2)," October 20, 1995.
Wolf Creek	4.2E-5/yr	6.3E-5/yr	No additional information was provided	Workshop Presentation by WOG, April 8, 1997.
Zion 1&2	4.0E-6/yr	4.8E-6/yr	Update included: - modeling and data changes - complete revision of HRA - several hardware and procedural improvements Update resulted in new dominant sequences	Commonwealth Edison Company, "Zion Individual Plant Examination (IPE) Response to NRC Staff Evaluation Report and Modified Zion IPE," September 1, 1995 Commonwealth Edison Company, "Commonwealth Edison Company Comments Regarding Draft NUREG-1560," February 14, 1997.

* The most recently reported values are reflected in this table.
** No new information was provided

APPENDIX C

PUBLIC COMMENTS AND NRC RESPONSES

ON DRAFT NUREG-1560

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C.1 Introduction

NUREG-1560, Volumes 1 and 2 were initially issued in October and November 1996, respectively as a draft report for public comment with the comment period ending May 9, 1997. At that time, notices were published in the *Federal Register* announcing the availability of the report and requesting comment (Ref. C.1). Distribution was made to over 500 people and organizations in the United States and abroad.

To assist readers of the document, a 3-day public workshop was held in April 1997 on the contents of draft NUREG-1560. A notice of this workshop was published in the *Federal Register* (Ref. C.2) and notification of the workshop was sent to all persons receiving the draft report. The workshop took place in Austin, Texas and was attended by representatives of the U. S. Nuclear Regulatory Commission (NRC) and their contractors, representative of the owner's groups, vendors, utilities and their contractors, consultants, and Federal and State agencies. A report summarizing the workshop was prepared and is available for inspection in the NRC Public Document Room (Ref. C.3). The report includes presentation material distributed at the meeting and summarizes the discussion periods during which questions were raised and responses provided. In addition, three sets of written comments were submitted at the meeting. These comments are included in Appendix C or the Workshop Summary Report. The authors and organizations submitting these comments are also listed in Table C.1 (Items #23-26).

In response to the request for comments, the NRC staff received 23 letters. The authors and organizations submitting these letters are listed in Table C.1. All letters received are available for inspection in the NRC Public Document Room.

Table C.1 Submitted comments on draft NUREG-1560.

Identification #	Organization	Author(s)	Date received by NRC
1	Commonwealth Edison Company	Thomas J. Maiman Executive Vice President (VP)	2-14-97
2	Niagara Mohawk	Martin McCormick, Jr. VP Nuclear Engineering	2-14-97
3	South Carolina Electric and Gas Company	Gary J. Taylor VP Nuclear Operations	2-17-97
4	----	Tony Spurgin	2-18-97
5	Centerior Energy	Lew W. Meyers VP	2-25-97
6	Duke Power Company	M.S. Tuckman Sr. VP Nuclear Generation	3-3-97
7	New York Power Authority	James Knubec Chief Nuclear Officer	3-4-97
8	Entergy Operations, Inc.	Jerrold G. Dewease VP Operations Support	3-7-97

Table C.1 Submitted comments on draft NUREG-1560.

Identification #	Organization	Author(s)	Date received by NRC
9	Illinois Power Company	Paul J. Telthorst Director, Licensing	3-7-97
10	Pacific Gas and Electric Company	Gregory Rueger Sr VP & General Manager	3-10-97
11	----	C.A Kukielka, Eric R. Jebsen	3-12-97
12	Carolina Power and Light Company	William Orser Ex.VP Energy Supply	3-14-97
13	PECO Nuclear	G.A. Hunger Director, Licensing	3-14-97
14	TU Electric	C.L. Terry Group VP	3-14-97
15	BWR Owner's Group	----	3-14-97
16	Westinghouse Owner's Group	Louis F. Liberatori, Jr. Vice-Chairman	3-25-97
17	Northeast Utilities Services Company	Sunil Weerokkody Supervisor, PRA	4-10-97
18	GPU Nuclear, Inc.	J.C. Fornicola Director, Licensing & Regulatory Affairs	4-29-97
19	Baltimore Gas and Electric Company	Charles H. Cruse VP Nuclear Energy	3-27-97 5-8-97
20	Public Service Electric and Gas Company	D.R. Powell Manager, Licensing & Regulation	5-9-97
21	IES Utilities, Inc.	John F. Franz VP Nuclear	5-9-97
22	Nuclear Energy Institute	Anthony Pietrangelo Director, Licensing Nuclear Generation	5-9-97
23	Environmental Protection Agency	T. Margulies	*
24	Virginia Power	K. Tuley	*
25	New York State Department of Health	J. Dunkleberger	*
26	NRC-IPE Workshop	**	*

* Written comments submitted at NRC-IPE workshop.
**Verbal comments discussed at NRC-IPE workshop.

In addition to these reviews and comments, as part of the normal review process, the staff discussed the approach and results of draft NUREG-1560 with the Advisory Committee on Reactor Safeguards on several occasions (Ref. C.4).

As discussed in Chapter 1 of this NUREG, the report is comprised of two volumes, with Volume 1 as a summary of the more detailed information contained in Volume 2. However, due to the nature of the comments received on the draft, some of the chapters were rearranged or renamed in the final report. Table C.2 shows the relationship of the draft report to the final report on a chapter by chapter basis. The comments received were reviewed and categorized according to the various chapters. Comments related to the "summary" chapter (from Volume 1) and the associated detailed chapter(s) (from Volume 2) are grouped together.

Table C.2 Relationship of draft NUREG-1560 to the final NUREG-1560

Volume 1 chapters			**Volume 2 corresponding detailed chapters**	
Final report	**Draft report**		**Final report**	**draft report**
1. Introduction	same	⇒	no corresponding chapter	no corresponding chapter
2. Impact of the IPE Program on Reactor Safety	same	⇒	9. Plant Vulnerabilities and Plant Improvements	same
no corresponding chapter	no corresponding chapter	⇒	10. Background for Obtaining IPE Results Perspectives	10. Background for Obtaining Reactor and Containment Design Perspectives
3. IPE Results Perspectives: Core Damage Frequency	3. Core Damage Frequency Perspectives	⇒	11. IPE Core Damage Frequency Perspectives	11. Reactor Design Perspectives
4. IPE Results Perspectives: Containment Performance	4. Containment Performance Perspectives	⇒	12. IPE Containment Performance Perspectives	12. Containment Design Perspectives
5. IPE Results Perspectives: Human Performance	5. Human Action Perspectives	⇒	13. IPE Human Performance Perspectives	13. Operational Perspectives
6. IPE Models and Methods Perspectives	6. IPEs with Respect to Risk-Informed Regulation	⇒	14. Perspectives on PRA Models and Methods Used in the IPEs	14. Attributes of a Quality PRA
				15. Comparison of IPEs to a Quality PRA
7. Additional IPE Perspectives	same	⇒	15. Safety Goal Implications	16. Safety Goal Implications
			16. Impact of Station Blackout Rule on Core Damage Frequencies	17. Impact of Station Blackout Rule on Core Damage Frequencies
			17. Comparison with NUREG-1150 Perspectives	18. Comparison with NUREG-1150 Perspectives
8. Overall Conclusions and Observations	same	⇒	no corresponding chapter	no corresponding chapter

All of the written comments sent directly to the NRC (Items 1-22 in Table C.1) and submitted at the workshop (Items 23-25 in Table C.1) together with all of the verbal comments provided at the workshop (Item 26 in Table C.1) have been addressed in the final version of NUREG-1560. The comments fell into three broad categories:

(1) A number of comments either were editorial in nature or address the accuracy of the information provided in draft NUREG-1560. For these comments, corrections were made to the text where appropriate. These comments are not reproduced in this appendix with staff response. The comments are available in the NRC Public Document Room.

(2) Some comments were observations in nature and did not appear to solicit a response nor seek a revision to the text of the report. These comments are also not reproduced in this appendix with staff response. The comments are available in the NRC Public Document Room.

(3) Other comments address insights, interpretations and perspectives drawn in the draft NUREG-1560. In some cases, the commentors were concerned that the conclusions were unsubstantiated. In other cases, commentors were concerned about policy implications. For these comments, summaries were developed that captured the concern and an NRC staff response to the comment is provided. These comments and associated responses are provided in the following sections. The specific comments are available in the NRC Public Document Room.

Some of the comments discussed in the following sections are more general in nature and applied to insights, interpretations, etc. discussed in more than one chapter of the report. Comments of this nature can, therefore, appear in several sections of this appendix. An attempt is made in each section to identify those comments that apply to other parts of the NUREG.

C.2 Chapters 2 and 9: Impact of the IPE Program on Reactor Safety

In addition to comments identifying factual errors in these chapters which were corrected, the following general comments were received. These comments and the NRC response are provided below.

1. **Comment:** Numerous erroneous claims of general applicability of vulnerabilities are made in the report. Implying generic applicability of vulnerabilities is inconsistent with the Individual Plant Examination (IPE) purpose which is to identify plant-specific vulnerabilities and cost-effective improvements. (Reference: see Table C.1, #8, 15, 20, 22)

   Response:

   It is true that the generic applicability of identified vulnerabilities cannot be ascertained. In addition, there is no consistent definition of vulnerability used in the IPEs. Further, variability in plant design and operation, as well as *different modeling assumptions*, can make a vulnerability unique to a particular plant. Therefore, statements regarding generic applicability of vulnerabilities have been rephrased in the NUREG. The purpose of presenting the vulnerabilities and associated plant improvements identified by the licensees is so that all of the licensees may benefit from considering these enhancements as means of improving the safety at their plant in a cost-effective manner.

2. **Comment:** Claims that plant improvements identified by one licensee could be implemented by other plants should not be made. Plant improvements should not be implemented without a full assessment of induced competing risks and the expenditure of resources required that may far outweigh any safety benefit gained. (Reference: see Table C.1, #15)

Response:

All statements about generic application of plant improvements have been rephrased in the NUREG. As with the identification of vulnerabilities, the purpose for discussing identified plant improvements is so that all licensees can benefit by considering their potential implementation at their plant to improve plant safety. A prudent evaluation by a licensee of the benefit of plant improvements identified by other plants would involve both cost-benefit and competing risk considerations.

3. **Comment:** Listing improvement implementation by the licensees as of the date of the IPE submittal is misleading because many plant changes have occurred since the initial IPE submittals. (Reference: see Table C.1, #1, 16)

   Response:

   NUREG-1560 represents a snapshot in time as far as risk and identified vulnerabilities and plant improvements (including their implementation). It is recognized that many licensees have updated their IPEs and the current status of identified plant improvements may be different than from what was reported in the original submittal. Updated plant improvement status reports are presented in Appendix B for those licensees who provided updated status information in response to the solicitation of comments on Draft NUREG-1560.

C.3 Chapters 3 and 11: IPE Results Perspectives: Core Damage Frequency

Many comments were received concerning the accuracy of the information provided in these chapters or the insights that were identified. Corrections were made to the text where appropriate. In addition, several general comments were provided on the content of this chapter. These comments and an associated response are provided below.

1. **Comment:** The reported core damage frequencies (CDFs) and dominant contributors do not reflect updated probabilistic risk assessment (PRA) results. Many utilities have updated their PRAs one or more times in response to plant design and procedure changes. In addition, many licensees have provided the NRC with revised IPE submittals some with extensive modeling changes and changes in the risk contributors and CDF. To correctly reflect insights from the IPEs requires consideration of supplementary submittals as well. (Reference: see Table C.1, #1, 12, 15, 22)

   Response:

   Because many plant PRAs are being constantly updated to reflect the current plant design and operation, it is not practical to constantly update NUREG-1560 to incorporate new insights. NUREG-1560 is, and will remain, a compilation of the calculated CDFs and insights obtained from the original IPE submittals. However, information from updated IPE submittals is provided in Appendix B.

2. **Comment:** In comparing the plants, the categorization of boiling water reactors (BWRs) solely by vintage, pressurized water reactors (PWRs) by nuclear steam supply system (NSSS) vendor, and Westinghouse PWRs by the number of loops is not appropriate and can lead to misinterpretation of results. It would be valuable to also look at the results based on a categorization of architect/engineer and/or builder and also age of plant to see if variations can be explained within each NSSS category. Further subgrouping of plants according to similar design characteristics (e.g., emergency core cooling system, ECCS, designs) could be possible. (Reference: see Table C.1, #16)

   Response:

   Early in the IPE Insights Program, the plants were grouped by architect/engineer and the IPE CDFs within and among these groups were compared. It was found that comparison of results on this basis was not productive because there is considerable design variability even among plants designed by the same architect/engineer. A decision was made to perform the analysis using plant groups based upon the NSSS vendor to account for basic NSSS design differences. The BWRs were further sub-categorized by vintage to account for differences in ECCS design. The Westinghouse plants were grouped according to the number of loops since the ECCS and other general plant features for the plants in each of these groups are generally the same (see Table 10.3). It is recognized that the balance of plant including support systems for plants in each of the designated groups can be different and skew any comparison of the results for a plant group. The NUREG identifies that these plant-specific features impact the results and draws the appropriate conclusions on the resulting insights. Finally, it is recognized that further subcategorization of plants according to a selected parameter could be made. However, variability in other parameters would likely impact that comparison. Because of this fact and also due to resource limitations, further subcategorization was not pursued.

3. **Comment:** The degree to which a search for variability associated with plant design differences has been made is questionable. The NUREG states that important design features, operator actions, and model assumptions all impact the variability in results. However, few model assumptions are identified. As is well known, substantial differences in PRA results occur because of balance-of-plant and support system design differences despite similarities in NSSS design. Therefore, it is judged that there is no basis to assert that the basis for observed variability is anything but dominated by plant differences in design, procedures, and training. (Reference: see Table C.1, #15)

Response:

Whether plant-specific design/operational differences or modeling assumptions are dominant factors in explaining the variability is not always obvious. However, it is believed that either or both can play a significant role in the variability for certain accident types. In many cases, a judgment is made in the NUREG on which is the dominant factor for an accident class for a plant group. The NUREG identifies that a significant amount of variability is due to support system and other plant-specific design/operational differences. Many of these design/operational differences are highlighted in the report. However, it is also clear that modeling assumptions play an important part in the variability. In some cases, because of limited documentation in the IPE submittals, it is not clear if the modeling assumption really reflects a design or operational difference. For example, many licensees did not credit an alternate coolant injection system because they did not perform an analysis of whether or not it would be successful. The neglect of the potential use of this system is a model assumption until it is shown that, because of plant-specific factors, such a system could not be used. For other cases, it is clear that a model assumption is being made. For example, many licensees assumed that the DC bus load shedding would always successfully occur during a station blackout.

4. **Comment:** The choice of success criteria has a major impact on the variability of the CDF results in a given category of plants. This is not mentioned in the NUREG. Some utilities working with smaller PRA vendors had more stringent success (i.e., conservative) criteria than others who worked with reactor vendors and had access to information that allowed for less conservative success criteria. Also, some larger utilities had the resources to perform the necessary analyses to establish a less conservative success criteria where other utilities did not have such resources and chose to use a conservative success criteria. (Reference: see Table C.1, #16)

Response:

The NUREG identifies where success criteria assumptions impact the variability of the calculated CDFs. As mentioned in the response to the previous comment, because of limited documentation in the submittals, it was not always clear if differences in success criteria were due to design differences or modeling assumptions. The basis for not crediting a system (and in some cases, for crediting a system) or for the operating requirements of a credited system (including support system requirements) were not always documented in the submittals. The CDF evaluation thus made no attempt to validate the differences in success criteria but simply reported its impact on the variability on the results. Also, Chapters 10 and 14 in the NUREG discusses the importance of success criteria to the results in general terms.

5. **Comment:** The NUREG should address the criteria used to determine what constitutes core damage. Many IPEs use core uncovery while others use a peak cladding temperature of 2200°F. This is important in that it impacts what equipment can be used to avoid core damage. (Reference: see Table C.1, #11, 15)

Response:

The impact of the definition of core damage on success criteria is discussed in general terms in Chapters 10 and 14. Specific impacts on the variability of the reported CDF definitions were not addressed because insufficient information was provided in the IPE submittals.

6. **Comment:** A discussion on how the component failure rates and the common cause failure rates impact the results is missing from the NUREG. This could be particularly important for assessing the importance of station blackout (SBO) since the reliability of on-site emergency AC power is critical. (Reference: see Table C.1, #16)

Response:

Because of the variability in the IPE modeling, it is not possible to always ascertain the impact of component failure rates and common cause failure rates. However, these factors were considered in establishing the parameters affecting the variability in the reported CDFs. Selected comparisons were made and, as discussed in Chapter 11, these failure rates were found to be important to the CDF variability. Also, based on a limited survey of data, Chapter 14 indicates that a wide variety of failure rates were identified in the IPEs for some components. This variability applies not only to plant-specific data but also to generic failure rates identified in the submittals.

7. **Comment:** Care must be taken when comparing the CDFs from transient events and from loss of coolant accidents (LOCAs). The IPEs approach the modeling of consequential LOCAs (e.g., reactor coolant pump, RCP, seal LOCAs or stuck-open power-operated relief valves or safety relief valves, SRVs) differently. Sometimes the CDFs from these events are reported in the transient contribution and sometimes in the small or medium LOCA CDF. It needs to be clearly stated how this is handled in NUREG-1560. (Reference: see Table C.1, #16)

Response:

It is true that there was considerable variability among the IPEs with regard to grouping sequences (for reporting) involving consequential LOCAs. However, the majority of the submittals reported sequences initiated by either a rupture or an inadvertent open SRV as LOCAs, and sequences with consequential LOCAS occurring after some other initiator as transients. This format was chosen for categorizing and reporting the results. For those IPEs that did not provide the results according to this format, an attempt was made to regroup the results to allow for comparison with the CDFs for other plants. However, in some cases, insufficient information was provided in the IPE submittal to distinguish the CDFs associated with these different accident sequences. In those instances, the licensee's reported results for transients and LOCAs were used directly.

8. **Comment:** It is not clear where special initiators fit into the CDF information reported in the NUREG. Generally loss of component cooling water and loss of service water can be important contributors to the CDF for PWRs due to the potential for an RCP seal LOCA. It would be advantageous to report the transient results in terms of CDF due to loss of decay heat removal and the CDF due to consequential LOCAs. (Reference: see Table C.1, #16)

   Response:

   It is agreed that it would be useful to separate the contributions from loss of decay heat removal and consequential LOCAs for the transient sequences. However, this information is not available consistently from the IPE submittals. Estimates were made from the reported information, whenever possible, and used in the report to identify relevant insights. The NUREG identifies that consequential LOCAs are important contributors to the CDF for many BWRs and PWRs.

9. **Comment:** The discussion on LOCAs should be directed at the ability of plants to mitigate small LOCAs. Overall, large LOCAs are not significant contributors to CDF. (Reference: see Table C.1, #16)

   Response:

   Significant contributions were observed from different sizes of LOCAs in different submittals. Therefore, it is not always true that large LOCAs are not significant contributors and that the discussion should focus on only small LOCAs. The NUREG discussion identifies what sizes of LOCAs dominate the LOCA contributions in each plant group and the reasons why.

10. **Comment:** A basis for the key perspective that PWRs with better feed-and-bleed capability generally have lower CDFs should be provided. There are many other plant design features and modeling methods that have a greater impact on CDF. (Reference: see Table C.1, #16)

    Response:

    The observation is made in the context of all PWRs. Within the Westinghouse plant groups, other factors besides feed-and-bleed capabilities are more important for explaining differences in transient CDFs and are discussed in the report. However, differences in feed-and-bleed capabilities are important when comparing across all PWRs because of the Babcock & Wilcox and Combustion Engineering plant design differences.

11. **Comment:** It is not clear from the information presented that the Westinghouse RCP seal LOCA model provides a lower contribution to CDF than the IPEs that used the NUREG-1150 model.

    Since this is very important to many plants, it is recommended that NUREG-1560 provide a detailed comparison of the two approaches. One of the dominating factors in the seal LOCA model is the probability of core uncovery occurring within the first hour. IPEs using the Westinghouse RCP seal LOCA model typically use 0.0283 and the NUREG-1150 model uses 0.0. The NUREG-1150 model does not consider any seal leakage for the first 90 minutes. From these facts it appears that the Westinghouse RCP seal LOCA model is more conservative. (Reference: see Table C.1, #16)

    Response:

    A comparison of the seal LOCA probabilities from the two models was not possible due to the unavailability of the reports documenting the Westinghouse model (with and without seal binding and popping open included). However, the Point Beach IPE and the response to questions concerning the Farley IPE did provide an opportunity to compare the core uncovery probability as a function of time for cases involving RCPs equipped with the old o-ring elastomer with the vessel either depressurized or not depressurized, and with the RCPs tripped. A comparison of the values from these curve fits with the core uncovery times calculated for identical cases for the Surry plant, as reported in NUREG-1150, Volume 3, is provided in Figure C.1. The curve fit is only valid over the time frame of 30 minutes to 8 hours. It should be noted that Point Beach is a two loop plant while Surry and Farley are three loop plants and thus the core uncovery time for a given leak rate could be different. However, since the reactor coolant system volumes for the plants are roughly scaled by the number of coolant loops, the core uncovery times for three plants for the same amount of leakage from each pump should not be substantially different. Thus, the core uncovery probability comparison in Figures C.1 provides a reasonable picture of the differences between the NUREG-1150 and Westinghouse seal LOCA models.

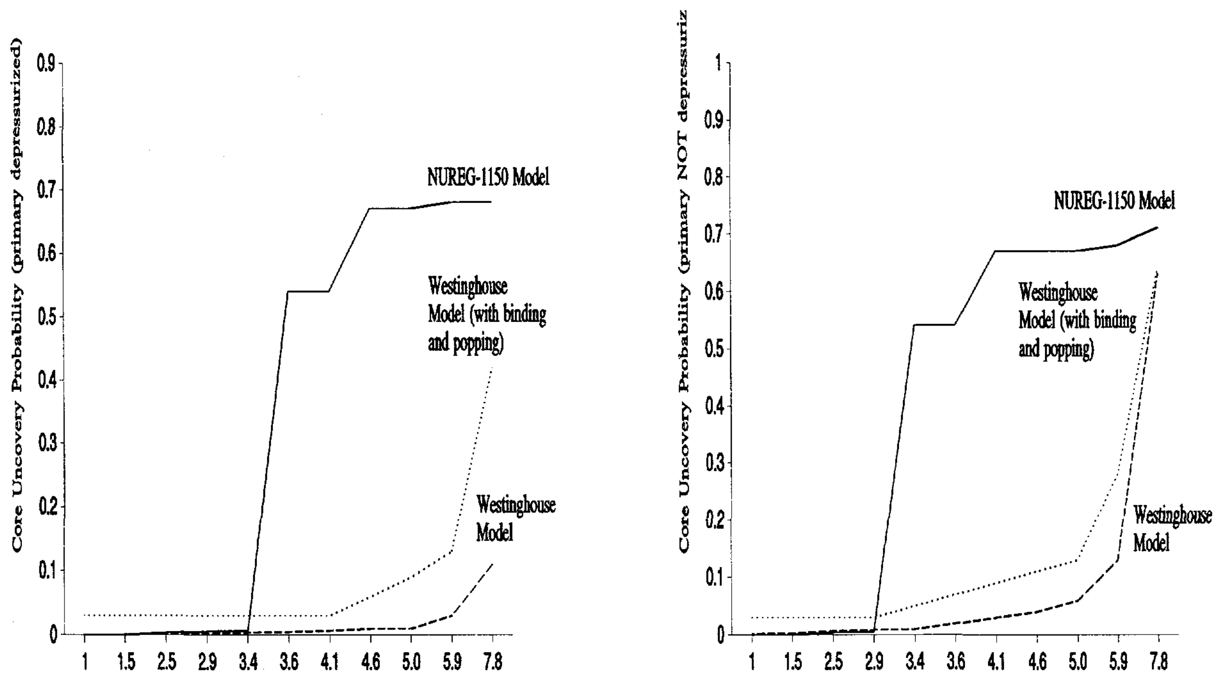

Figure C.1 Comparison of NUREG-1150 and Westinghouse seal LOCA models — old o-ring elastomer.

Figure C.1 indicates that all three models predict small probabilities of leaks and core uncovery for early times (less than about 3 hours). Because of this, differences between the three models do not have a significant impact on CDF for this early time period. However, for later times, the differences are more significant. The Westinghouse models generally predict much smaller probabilities for core uncovery for time periods greater than approximately 3 to 3.5 hours, particularly for cases where the vessel is depressurized. For scenarios where the vessel is not depressurized, however, the probabilities predicted by the Westinghouse models rise sharply at about 8 hours, so that the three models give similar probabilities at that time.

The fact that seal LOCAs occur in all three models does not mean that the impact on the CDF will be the same in both cases. As noted earlier, none of the models result in a significant contribution to CDF in the first three hours. However, unlike the Westinghouse models, the NUREG-1150 model can result in significant contributions to CDF based on core uncovery in the 3 - 8 hour time frame. For example, in this time frame during a station blackout, the core is likely being cooled by auxiliary feedwater, given that battery power is still available. Therefore, without a seal LOCA, core damage would not be expected during this time frame. For times past 8 hours, all three models predict a high probability of a seal LOCA leading to core uncovery. However, for these longer times, battery depletion would have occurred at most Westinghouse plants, leading to loss of heat removal and boiloff. Therefore, if AC power recovery does not occur, core damage will result whether or not a seal LOCA is present. In this situation, the station blackout CDF is not affected by small seal LOCAs that would result in core uncovery at times greater than 8 hours. The precise impact of the model differences is plant-specific, depending on battery depletion times and AC power recovery alternatives. Similar impacts occur for non-station blackout scenarios (e.g., loss of component cooling water events) where the seal leakage rate impacts the time available for other recovery actions such as arranging alternate charging pump cooling.

The documented NUREG-1150 seal LOCA model indicates no seal failure prior to 90 minutes. However, after most of the IPEs were completed, an error in the NUREG-1150 model was identified which indicates that there should be some probability of seal failure immediately after loss of seal cooling. Thus, the contribution of RCP seal LOCAs in the IPEs that utilized the NUREG-1150 model is likely underestimated. An evaluation for the NUREG-1150 study for the Sequoyah plant indicates that the seal LOCA contribution was underestimated by 18% (corresponds to a core damage frequency of 7.7E-7/yr). Accounting for this error would slightly widen the difference between the Westinghouse and NUREG-1150 models.

12. **Comment:** The NUREG discusses uncertainty associated with the Byron-Jackson N-9000 seals and *"infers that the IPEs* [for plants with these pumps] *are suspect in their RCP seal LOCA conclusions."* Details concerning this technical issue have been provided to the NRC in various forms in the past. Please modify the NUREG to reflect the technical information provided and remove the inference that the IPEs are suspect in their RCP seal LOCA conclusions. (Reference: see Table C.1, #8)

Response:

NUREG-1560 reflects the information provided in the IPE submittals which indicate that the contribution from RCP seal LOCAs is generally small for plant with Byron-Jackson pumps. The NUREG reiterates the statements made in the submittals that there is little or no potential for seal LOCAs in these plants if the RCPs are tripped. The submittals cite the design of the pumps some limited analyses, test, and actual experience as the basis for this argument providing some references. No judgement is made in NUREG-1560 about the accuracy of the RCP seal LOCA modeling for these plants based on the information in the submittal. However, the potential for RCP seal LOCAs in these plants is still being reviewed as part of NRC's Generic Safety Issue 23. The information cited in the submittals as the basis for the RCP seal LOCA modeling is being examined as part of the resolution of this issue.

13. **Comment:** The reported CDFs have been rounded to one significant figure. The NUREG should report the actual CDFs reported in the IPEs. (Reference: see Table C.1, #8)

Response:

A decision was made to report the CDFs to one significant figure (to provide consistency) and are based on the actual values reported in the IPE submittals.

C.4 Chapters 4 and 12: IPE Results Perspectives: Containment Performance

1. **Comment:** Conditional containment failure probability (CCFP) is not a good measure of safety performance. The use of conditional measures implies an independence between the systems which prevent core damage and the systems which prevent containment failure which is part of the design of the current generation of light water reactors. Plants with relatively higher CCFPs are not necessarily less safe than those with relatively lower CCFPs. The measures which impact public safety are related to the frequency of releases from the containment. (Reference: see Table C.1, #11, 12, 22, 23)

   Response:

   One of the main objectives of the chapters in NUREG-1560 related to containment performance is to obtain perspectives on the performance of the various containment types independent from other plant features. For this purpose, the CCFP is a useful parameter since it decouples containment failure from core damage frequency. This was also recognized by the majority of licensees since CCFPs are reported directly in most of the IPE submittals. Ideally, the comparison of containment performance among different IPEs would be accomplished by comparing CCFPs for individual plant damage states. However, such a comparison is not possible since the definition of the plant damage states was left to the individual analyst and thus varies from IPE to IPE. NUREG-1560 also recognizes that the probability of containment bypass is not a measure of containment performance in the same way that isolation or structural failure of the containment is. Therefore, the NUREG separates the conditional probabilities of containment bypass and containment "failure" when making comparisons. The importance of containment failure frequency is acknowledged in Chapter 12 of the NUREG where comparisons of containment failure frequencies as well as release frequencies are also presented. The NUREG does not draw conclusions or make implications regarding overall plant safety based on CCFPs. Containment failure probabilities are used only to compare the containment performance among plants with the same type of containment and among different containment types. For this purpose the CCFP is the best suited parameter.

2. **Comment:** The report utilizes at least five different figures of merit in characterizing containment performance. It is never clear which figure is most appropriate or why. The figures include: total conditional containment failure probability, conditional probability of various containment release types (bypass, early failure, late failure), frequency of bypass and early release, conditional probability of "significant early release," and frequency of releases with the potential to cause early fatalities. (Reference: see Table C.1, #22)

   Response:

   NUREG-1560 uses various parameters related to containment performance in different chapters of the report depending on the purpose of the comparisons to be made and the perspectives to be obtained. There is no single "most appropriate" containment performance figure of merit for the whole report, nor should there be. Those parameters which best served to illustrate the points to be made for the issues at hand were chosen in different sections of the report. Total conditional containment failure probability is not used in the NUREG. For purposes of obtaining perspectives on containment performance, conditional probabilities of containment bypass, the CCFPs for early and late failure are used in Chapters 4 and 12. Conditional probabilities of significant or large early release, defined as early releases where releases of Cs, I and Te exceeded 0.1 of core inventory, are also compared in these chapters since this type of release was singled out in many IPE submittals. Finally, frequencies of early release from bypass and early containment failure were used in Chapters 7 and 16 since this parameter was the one which allowed an indirect comparison of the IPE results with the safety goals.

3. **Comment:** While there have been some mis-applications of MAAP, any implication that the MAAP code is inadequate is wrong. It is misleading to state that MAAP does not have a comprehensive treatment of severe accident phenomena. A more problematic item involves the utilities which did not properly apply MAAP and/or relied on the industry position papers. (Reference: see Table C.1, #8, 11, 12, 22)

   Response:

   MAAP as well as other system level codes do not cover the range of postulated severe accident phenomena (e.g., steam explosions, direct containment heating, shell melt-through, hydrogen detonation). This is what is meant by the statement that the MAAP code does not have a comprehensive treatment of severe accident phenomena. The EPRI report on MAAP acknowledges *"one should recognize that MAAP cannot and does not contain detailed models for all phenomena."* As noted above, other system level codes share this limitation, and this is one reason why the IPE guidance called for proper sensitivity studies to be conducted as part of the Level 2 analysis. In some cases MAAP was applied by the IPE analysts in a way that did not follow industry recommended guidelines. NRC noted *"...the adequacy of the MAAP 3.0B code for use in the IPEs..."* but also stated that *"licensees...bear the burden of proof that they have applied the code properly, and that they meet the intent of the IPE generic letter."* Regarding the industry position papers, their application in an IPE to qualitatively dismiss a number of accident progression phenomena, without any sensitivity considerations, or without any understanding of the uncertainty associated with the different phenomena, is not in line with the intent of Generic Letter 88-20. This approach was less helpful in fostering a licensee's appreciation and understanding of severe accident behavior than a proper application of MAAP.

4. **Comment:** Results are presented by reactor and containment type and NSSS. It would be valuable to also look at the architect/engineer or builder to explain the variation in reported results. (Reference: see Table C.1, #16)

   Response:

   Early in the IPE Insights Program a decision was made to group the containment performance results under the five common containment classes used in the United States. Containment response to severe accidents has been found to correlate to these five containment classes as illustrated in the NRC's Containment Performance Improvement program. In discussing containment performance perspectives, NUREG-1560 identifies those architect/engineer specific containment construction features which play a significant role in the IPE analysis, as reported in the IPE submittals. These features include the containment material, layout of reactor cavity, and location of sumps and drain lines.

5. **Comment:** It is judged that there is no basis in NUREG-1560 to assert that the observed variability in the IPE results is anything but dominated by plant differences in design, procedure, and training. (Reference: see Table C.1, #15)

   Response:

   In discussing containment performance perspectives, NUREG-1560 identifies the plant specific differences described in the IPE submittals which lead to some of the variability in the reported results. However, it is clear that modeling assumptions also play an important role in the observed variability in containment performance. Assumptions regarding the amount and composition of core material exiting from the reactor vessel, the coolability of this debris, and the pressure and temperature rise in the containment due to core debris dispersal are examples of modeling assumptions which had a significant influence on the assessment of containment performance. Other assumptions include the likelihood of in-vessel recovery of the accident, including the likelihood of retaining the core debris in the reactor vessel via external cooling of the vessel.

6. **Comment:** It would seem prudent to avoid misinterpretations by providing the specific NRC assumptions used in extrapolating IPE submitted words to the construction of the comparisons among plant results in NUREG-1560. These assumptions would include:

   - What the relationship of containment vent treatment is to the CCFP, the early releases, and other measures of risk;
   - what the correlation is between each IPE result for early and late releases and their definition of "early" and "late";
   - how the assignment of multiple containment failure modes affects the assignment of the allocation of failure modes in comparisons (e.g., shell melt-through following wetwell failure); and
   - defining the treatment of dynamic failure modes and their associated failure locations as it relates to inferences about failure locations and timing. (Reference: see Table C.1, #15)

   Response:

   There exists detailed discussion in the appropriate sections of Chapters 4 and 12 of NUREG-1560 on:

   - how venting was grouped to the different containment failure modes.
   - how "early" and "late" was defined in the comparison of failure modes and releases.
   - how multiple containment failure modes were treated as they were reported in the IPE submittal (i.e. whichever failure mode was considered dominant in the submittal base case results was the one used in NUREG-1560).
   - The above comment on the treatment of dynamic failure modes is not clear, and no further clarification was provided at the workshop; consequently, no changes were made to NUREG-1560.

C.5 Chapters 5 and 13: IPE Results Perspectives: Human Performance

1. **Comment:** It is stated in the report that in most cases there is little evidence that the human reliability analysis (HRA) quantification method per se has a major impact on the results. This seems to imply that *"the impact of HRA on PSA can best be described as indeterminate"* or *"that the HRA seems to have little effect on the results of the PRA."* If this is the case, why are the HRAs identified as important shortcomings of the IPEs and why is the quality of the HRAs a concern. (Reference: see Table C.1, #8, 11, 12, 15, 21)

   Response:

   The interpretation that *"the impact of HRA on PSA can best be described as indeterminate"* or that *"the HRA seems to have little effect on the results of the PRA"* is not what was meant. How and how well the HRA method is applied and the resulting human error probabilities (HEPs) clearly have significant impacts on the results of the PRA. Thus, it is for this reason that concern is raised in the NUREG about the "quality" of the HRAs performed by the different licensees. The statement that *"in most cases there is little evidence that the HRA quantification method per se has a major impact on the results,"* was meant to imply that the HRA results from the different IPEs did not in general appear to vary directly as a function of the particular or "nominal" HRA method used, e.g., the Technique for Human Error Rate Prediction versus the Success Likelihood Index Methodology versus the Human Cognitive Reliability model. The variability in results appeared to be more a function of how or how well the HRA methods were applied or the impact of plant-specific characteristics, as opposed to which nominal HRA method was used. Due the confusion caused by the statement and the fact that the direct impact of the nominal method per se is difficult to evaluate given the many other relevant factors, the statement was deleted from the final NUREG. Additional clarification regarding the quality of the HRAs performed for the IPEs is provided below in the response to Comment #2.

2. **Comment:** In spite of the assertion in the report that *"it appears that there are reasonable explanations for much of the variability in HEPs and in the results of the HRAs across the different IPEs,"* it is also asserted that because *"many of the licensees failed to perform high-quality HRAs, it is possible that the licensees obtained HEP values that are not appropriate for their plants."* These statements appear to be inconsistent. Moreover, others sections of the report indicate that not all of the variability in HEPs could be explained. Please provide clarification on what appears to be inconsistent statements and address the assertion that *"many of the licensees failed to perform high-quality HRAs."* (Reference: see Table C.1, #1, 2, 8, 11, 12, 22)

Response:

Confusion arose regarding the implication or meaning of the significant variability in HEPs that was identified for selected human actions across plants, particularly in terms of the quality of the HRAs. Figures displaying the HEPs for several events (e.g., manual depressurization during transients) were presented in the report and discussions of the reasons for the variability were provided. Many of the comments received from licensees on this topic attempted to defend the variability on the basis of the numerous reasonable factors that would lead to the variability. That is, the values across plants may have been developed on entirely different bases. For example, different plants have different system characteristics and may have different procedures. Initiator and sequence-specific factors and dependencies will also lead to variability in HEPs. Moreover, some plants only used "screening values" in modeling some of the examined events. On the basis of these and other factors, the commentors indicated that such variability would be expected.

This conclusion is, at least in part, one point the staff was trying to make and which was stated in the report. That is, there are *"reasonable explanations for much of the observed variability in HEPs across plants."* In other words, the rather striking degree of variability, in at least nominally similar human actions, is based to some extent on valid differences. From this perspective it can be argued that the licensees attempted to consider relevant factors in obtaining the HEPs for operator actions and that the results of the HRAs performed by the different licensees were generally consistent and therefore useful. In fact, the staff does not in general disagree with this conclusion.

However, another conclusion reached by the staff and documented in this report was that not <u>all</u> of the variability in the examined HEPs was easily explained. That is, after "acceptable" reasons for variation were considered, there still appeared to be some degree of unexplained variation the HEPs (see Chapter 13). While some of this variation would be expected due to the lack of precision in existing HRA methods, it is also possible that some of the variation was due to factors such as analyst biases, invalid HRA assumptions made by analysts performing the HRAs, or superficial HRA analyses that failed to adequately examine and model the potential for human error (e.g., through careful consideration of plant-specific performance shaping factors (PSFs), consideration of dependencies, use of simulator exercises, etc). Due to the limited information provided in many submittals on the derivation of particular HEPs, it is difficult to determine the extent to which inappropriate factors actually influenced the derived HEPs. However, examinations of the submittals during the project indicated that not all licensees performed quality HRAs. That is, not all licensees applied the existing HRA methods as well as they could have. For example, they did not always consider dependencies, appropriately assess the impact of time availability, or carefully consider plant-specific PSFs. Some failed to model pre-initiator actions and others did not conduct simulator exercises or perform walkdowns and timing of operator actions to be conducted outside the control room, etc. The conclusion that not all licensees conducted high-quality HRAs is further documented in some of the staff evaluation reports (SERs) that have been issued on the submittals. Some submittals indicated as having met the intent of Generic Letter 88-20 were found to have various weaknesses that could have influenced the HEPs obtained for particular events.

While the degree of consistency in HEPs obtained for similar human actions in similar contexts suggests that in general the HRA results from the IPEs were useful in terms of meeting the intent of Generic Letter 88-20, it should be further noted that even when reasonable consistency exists, it is not necessarily the case that all the HEPs calculated by a particular plant were realistic and valid for that plant. As noted in Chapters 5 and 13, reasonable consistency can be obtained in HRA without necessarily producing valid HEPs. An HEP is only valid to the extent that a correct and thorough application of HRA principles has occurred. For example, if a licensee simply assumed (without adequate analysis) that their plant is "average" in terms of many of the relevant PSFs for a given event, but then does appropriately consider the time available for the event in a given context, the value obtained may be similar to those obtained for other plants with similar time frames for the event. Yet, the resulting value may be optimistic or pessimistic relative to the value that would have been obtained if the licensee had conducted a detailed examination of the relevant plant-specific factors. Thus, while the degree of consistency obtained by the licensees is encouraging regarding the ability to compare the results of the IPEs, and while many licensees performed excellent HRAs, the fact that some licensees did not perform as thorough HRAs as possible given the state-of-the-art in HRA at the time, means that the results are not as good as they might have been. It does not mean that individual licensees and the industry in general did not obtain important information from performing the IPEs.

3. **Comment:** By questioning the quality of the HRAs performed for the IPEs, NUREG-1560 seems to imply that the licensees should have attempted to extend the state-of-the-art in HRA in order to obtain quality results. (Reference: see Table C.1, #8, 11, 21)

Response:

The staff believes that the state-of-the-art in HRA at the time of the IPEs was adequate for the intent of Generic Letter 88-20. The shortcomings related to the HRAs performed for the IPEs were in how the existing methods were applied, rather than the methods themselves. Of course, this position does not imply that improvements are not needed in HRA, but rather that useful results can be obtained with thoughtful and thorough applications of existing methods.

4. **Comment:** The NRC needs to initiate a number of policy and research activities to address shortcomings both in the NRC's attitudes and strategies for ensuring that the licensees maintain safe plants and in the development and use of PRA and HRA methods and techniques. These activities (summarized) include establishing a regulatory attitude that encourages the licensees to be pro-active rather than reactive (to the NRC) in ensuring plant safety, encouraging more thorough and realistic HRAs, supporting the development of multiple new approaches to HRA (which include more effective use of simulators), reevaluation of the real contribution of common cause to risk, reevaluating the use of Bayesian updating during "period of rapid changes in maintenance," and investigating the impact of management and organizational factors on plant safety. (Reference: see Table C.1, #4)

   Response:

   The author (of the comments summarized above) acknowledged that the *"comments are not just on the NUREG document itself, but are also directed towards some overall aspects of PRAs and HRAs."* However, none of the comments appear to address the NUREG itself. Nevertheless the NRC does currently have programs addressing each of the issues raised by the author, e.g., development of improved HRA methods and consideration of the impact of management and organizational factors on plant safety. Further, the NRC staff has reviewed the comments and will consider them in future directions of research.

C.6 Chapters 6 and 14: IPE Models and Methods Perspectives

Several comments were received expressing technical disagreement with some of the information provided in these chapters. The text was revised where appropriate. In addition, several general comments were provided on the content of these chapters. These comment and associated responses are provided below.

1. **Comment:** Numerous comments were received on the description of a "quality" PRA in Chapters 6 and 14 and on the comparison of the IPEs to a quality PRA in Chapters 6 and 15 of the draft NUREG. Several commentors felt that these chapters were inappropriate for NUREG-1560 and that they should be deleted from the final report. This recommendation was largely driven by the assumption that the attributes of a "quality" PRA were intended to be standards or requirements and that all the attributes had to be met prior to using PRAs in future risk-informed regulatory activities. Given that some commentors felt that the PRA attributes were too demanding, overly prescriptive and beyond the current state-of-the-art, it follows that if they were assumed to be requirements then they could be interpreted as a significant burden on the industry. Several comments emphasized that the scope and attributes of a PRA to be used for risk-informed regulatory activities should be commensurate with the application. This implies that PRAs with significantly less attributes and of more limited scope than the PRA described in NUREG-1560 would be acceptable for risk-informed applications. Other commentors stressed that any applications of the PRA attributes in NUREG-1560 to the creation of an industry standard should be viewed as developmental in nature. An industry-wide standard for PRA quality should be based on a broader and more deliberate development effort that involves practitioners from various

organizations. (Reference: see Table C.1, #1, 2, 8, 9, 15, 16, 20, 22 and 26)

Response:

Chapters 6, 14 and 15 of draft NUREG-1560 have been significantly revised for the final report. Specifically, Chapters 14 and 15 have been replaced with a new Chapter 14, and references to the use of the IPEs in risk-informed regulation have been removed. Chapters 6 and 14 in the final report summarize PRA characteristics and state that they:

- are not "standards" nor do they represent regulatory guidance.
- are included only as a benchmark in order to draw perspectives on the models and methods used in the IPEs.
- do not define the needed quality or scope of the PRA elements needed for a particular regulatory application.

2. **Comment:** Several comments were related to the following statement in draft NUREG-1560, *"...and other utility personnel are excluded from the peer review team."* This statement was interpreted by some commentors as implying that no employees of any utility can serve as a peer reviewer. (Reference: see Table C.1, #1, 8, 12, 15, 16, 20 and 22)

   Response:

   This interpretation was not intended. The statement was included simply to indicate that it would be inappropriate for utility staff to be part of the PRA peer review team for plants owned and operated by their utility. NUREG-1560 has been revised accordingly.

C.7 Section 7.1 and Chapter 15: Safety Goal Implications

Several comments were received expressing technical disagreement with some of the information provided in these chapters. The text was revised where appropriate. In addition, several general comments were provided on the content of these chapters. These comments and associated responses are provided below.

1. **Comment:** The concern is that the results reported in the original IPE submittals are not current and could be misleading when compared to the Safety Goals. For example, several plants identified in Chapters 7 and 16 in Draft NUREG-1560 (Chapter 15 in Final NUREG-1560) as potentially approaching the early fatality quantitative health objective (QHO) have subsequently updated their PRAs with significant reductions in CDF and large early release frequency, LERF (including Browns Ferry, Beaver Valley and Palo Verde). (Reference: see Table C.1, #22, 25 and 26)

   Response:

   NUREG-1560 has been revised to clarify that the perspectives on the safety goal are based on the original IPEs/PRAs which may have subsequently changed. However, the results quoted in NUREG-1560 will not be revised. New information obtained by the staff will be included in NUREG-1560 (see Appendix B). In the case of the safety goal comparisons if any of the plants that were identified as approaching the early fatality QHO submit revised results, this will be noted in Chapter 7 and 15 and the reader will be directed to the appendix.

2. **Comment:** Inferences that a few plants may approach the early fatality health objective based on a comparison of the IPE and NUREG-1150 results may not be valid. Additional insights gained from the containment performance evaluations and recent research in the area may

lead to different conclusions than the NUREG-1150 analyses. (Reference: see Table C.1, #16)

Response:

NUREG-1560 has been revised to clarify that NUREG-1150 containment results were not used to link the IPE results to the safety goals. For early fatality risk, a two step process was used. In the first step, the frequencies of early containment failure and bypass were obtained from the IPEs and plants with low frequencies ($<10^{-5}$/ry) were screened out from further consideration. For the remaining plants, the frequencies of source terms with relatively large release fractions (>0.03 Cs, I, Te) were obtained. The source term frequencies were then adjusted for population and compared to the goal.

3. **Comment:** There is an implication in the report that the only way a comparison can be made to the "Safety Goal" is to have a Level 3 PRA. Such a PRA was never mandated, requested or suggested by the NRC and there are a number of ways to compare to the Safety Goal other than having a Level 3 PRA. The NUREG could address how the NRC and industry (there are several EPRI documents and other papers, positions and reports) have defined or linked the NRC "Safety Goal" in terms of Level 1 and 2 surrogate indicators. (Reference: see Table C.1, #15, 21)

   Response:

   The approach used by the staff in Chapters 7 and 15 of NUREG-1560 was based on using Level 1 and 2 surrogate indicators to link the IPE results to the safety goals. The wording in Section 6.4 has, therefore, been changed to make it clear that a Level 3 PRA is not the only way to make a comparison to the safety Goals.

4. **Comment:** One comment stated that conclusions based on using the IPE results for comparisons to the QHOs of the safety goals must be carefully qualified. The purpose of the IPEs was not to define absolute risk levels, but rather to identify plant severe accident vulnerabilities. Consequently, the safety goal computations performed by the staff (described in Chapters 7 and 16 of draft NUREG-1560) are not an adequate technical basis on which such a conclusion can be drawn. In a related comment, SECY-90-104 was quoted, *"based on the significant additional resources that would be required to make a meaningful comparison of the IPE results with the safety goal policy statement and the potential problems associated with using the as-submitted IPE data, the staff recommends that no direct comparisons be made unless the IPEs are reviewed to a greater level of detail than currently planned."* As the commentor believes that a review of greater detail did not occur, it was recommended that the direct comparison of IPEs to the Safety Goals in Chapters 7 and 16 be removed from the final NUREG. (Reference: see Table C.1, #19, 22)

   Response:

   The final version of NUREG-1560 has been revised to clearly describe the limitations of the approach used to compare the IPE results to the safety goals and subsidiary objectives. However, the use of Level 1 and 2 indicator (CDF and LERF) as surrogates for the safety goals is consistent with recent industry positions (refer to Comment #3 in Section C.7 above) and consistent with the guidance provided by the NRC for use of PRAs in risk-informed regulatory applications (Ref. C.5). The manner in which the IPE results are compared to the safety goals is consistent with the "Integration Plan for Closure of Severe Accident Issues," SECY-88-147 and also consistent with the recommendations of SECY-90-104, namely, *"....indirect comparison of the IPEs and other available PRAs with the Safety Goals, focusing on the insights gained and the adequacy of regulations, is planned."* The SECY further recommends that the *"staff evaluate the IPE results as a whole and summarize any conclusions and recommendations for the*

Commission at the completion of the IPE process."

5. **Comment:** Several verbal and written questions were received at the workshop related to the appropriateness of the current safety goals and the manner in which comparisons were made to these goals. (Reference: see Table C.1, #23, 26)

   Response:

   The appropriateness of the current safety goals is a policy issue and outside the scope of NUREG-1560. The use of Level 1 and 2 indicators as surrogates for the safety goals is consistent with the staff's guidance provided in the recently published regulatory guides (Ref. C.x).

6. **Comment:** The definition of an early release, particularly a large early release, and the time available for effective evacuation after declaration of a general emergency appears to be arbitrary. Consideration of the accident timing, the site, and the impacts on evacuation (such as an SBO) need to be considered. (Reference: see Table C.1, #25)

   Response:

   A unique definition of a large early release was not provided in NUREG-1560. A large early release is defined in the staff's regulatory guides (Ref. C.x) on the use of PRA in risk-informed regulation. Numerical objectives for the frequency of a large early release are also provided in those documents. The frequencies of early containment failure and bypass were used in NUREG-1560 to screen out plants with low frequencies. The frequency of source terms with relatively large release fractions were then examined in more detail to estimate the potential early health effects. The assumption was made that these releases occur prior to effective offsite evacuation. This assumption could overestimate the potential for early health effects.

C.8 Section 7.2 and Chapter 16: Impact of Station Blackout Rule on Core Damage Frequencies

Several comments were received expressing technical disagreement with some of the information provided in these chapters. The text was revised where appropriate. In addition, several general comments were provided on the content of these chapters. These comment and associated responses are provided below.

1. **Comment:** Evaluation of the SBO rule would benefit from a review of the results by Architect/Engineer and not just by reactor type. (Reference: see Table C.1, #16)

   Response: As is discussed in the response to similar comments on Chapters 3 and 11 (and the report in general), early in the IPE Insights Program the plants were grouped by architect/engineer and the IPE CDFs within and among these groups were compared. No strong correlation with the architect/engineer was found because there is considerable design variability even among plants designed by the same architect/engineer. A decision was made to perform the analysis using plant groups based upon the NSSS vendor to account for basic NSSS design differences. The BWRs were further sub-categorized by vintage to account for differences in ECCS design. The Westinghouse plants were grouped according to the number of loops since the ECCS and other general plant features for the plants in each of these groups are generally the same (see Table 10.3). It is recognized that the balance of plant including support systems for plants in each of the designated groups can be different and skew any comparison of the results for a plant group. The NUREG consistently identifies that these plant-specific features impact the results and draws the appropriate conclusions on the resulting insights. Finally, it is recognized that further subcategorization of plants according to a selected parameter could be made. However, variability in other parameters would likely impact that comparison. Because of this fact and also due to resource limitations, further subcategorization was not pursued.

C.9 Section 7.3 and Chapter 17: Comparison with NUREG-1150 Perspectives

Several comments were received expressing technical disagreement with some of the information provided in these chapters. The text was revised where appropriate. In addition, several general comments were provided on the content of these chapters. These comment and associated responses are provided below.

1. **Comment:** Chapter 18 in Draft NUREG-1560 (Chapter 17 in Final NUREG-1560) presents a comparison of NUREG-1150 results with IPE results as a whole. A more interesting comparison would be between the individual NUREG-1150 results and the corresponding IPEs. This would provide a more detailed information on specific modeling issues. (Reference: see Table C.1, #2)

   Response:

   Section 7.3 indicates that the focus of NUREG-1560 is on comparing global perspectives discussed in NUREG-1150 with the overall results of the IPEs. A plant-specific comparison between NUREG-1150 and the applicable IPE analyses are provided in the individual SERs on the five IPEs. Chapter 17 in the Final NUREG-1560 has been revised to clarify the scope of the comparison in NUREG-1560 and to note that plant-specific comparisons may be found in the SERs.

C.10 Chapter 8: Overall Conclusions and Observations

Some general comments concerning the content of Chapter 8 were received from several organizations and individuals. Responses to these comments are provided below.

1. **Comment:** Due to the nature of the IPE process requested in Generic Letter 88-20 (a search for vulnerabilities, not characterization of absolute risk), the applicability of the IPE results for regulatory follow up activity should be limited. Section 8.2.4 states that the NRC staff plans follow-up activities to determine if additional regulatory actions are warranted for plants with relatively high CDFs or CCFPs. NUREG-1560 does not consider revised CDF and CCFP values provided to the NRC, which in some cases, are substantially different than the original IPE submittal values. Consequently, use of the IPEs for comparison to safety goals, identification of "outlier" plants, and for direction of inspection and follow-up activities should be minimized. Such actions have the potential to lead to ineffective use of NRC staff and utility resources in pursuing areas which are known to be outdated. The NRC staff should evaluate these changes in the plant CDF and CCFP values before planning follow-up activities. (Reference: see Table C.1, #20, 22)

   Response:

   The IPE results and insights provide a useful source of information for identifying areas where follow-up activities might be warranted. The information contained in NUREG-1560, however, is merely a starting point and is by no means the sole basis for regulatory decisions. Before any plant-specific actions are taken, the best available information will be considered, including any revisions to the original IPE submittals, recognizing that most of the newer information has not yet received staff review. Further, any proposed regulatory actions are subject to the Backfit Rule as described in 10CFR50.109.

2. **Comment:** The NRC staff's approach in looking at CDF and CCFP as independent factors is incorrect. It assumes the existence of either a high CDF or high CCFP is evidence on its own of a potential concern. In reality, the two factors should be looked at together. They are each a part of the overall input to risk, which should be the figure of merit (the CDF/CCFP criteria do not have any established technical connection to the QHOs of the Safety Goal). (Reference: see Table C.1, #8, 22)

   Response:

   The major objectives of the IPE Insights Program are outlined in both the Forward and Introduction of NUREG-1560. For at least one of those objectives (i.e., providing perspectives on plant feature and assumptions that play a role in the estimation of CDF, containment performance and human performance), it is useful to look at CDF and CCFP separately. The use of these parameters in NUREG-1560 does not imply that a high value for either parameter alone is a potential concern or will be the basis for regulatory decisions. Instead, the use of these parameters allows the staff to focus individually on the Level 1 and Level 2 analyses performed for the IPEs, thereby accomplishing the objectives noted above.

3. **Comment:** Concerning any follow-up regulatory activities, it's suggested that the investigation and regulatory considerations not be limited just to the high CDF or CCFP issues. Areas where the risk impact is small and the safety benefit is not appreciable should also be investigated for reduced regulatory burden. (Reference: see Table C.1, #6, 16)

   Response:

   The primary focus of the NRC is to assure the safety of the public. Therefore, it is natural that the NRC tends to be more concerned with eliminating vulnerabilities and reducing risks than with reducing burden. However, the latter objective is desirable and the NRC encourages the industry to submit requests for reduced regulatory burdens in areas where they believe that risks are low and substantial cost savings can be achieved.

4. **Comment:** The discussion of the Maintenance Rule says it is acceptable to use the IPEs to determine risk significant systems. However, this is not compatible with the findings about the usefulness of the IPEs for risk-informed regulation. Likewise, the NRC implies that for inspection purposes the IPEs are adequate for them to target areas for plant-specific inspections but NUREG-1560 states that the PRAs are only adequate to identify dominant accident sequence types and their relative importance. This seems inconsistent. Furthermore, the NRC seems to be attempting to use PRA information in a selective manner, where it serves their purposes. (Reference: see Table C.1, #22)

   Response:

   References to the use of the IPEs in risk-informed regulation have been removed from the final version of NUREG-1560. Issues related to the quality and scope of PRAs needed for risk-informed regulation are discussed in the staff regulatory guides, and standard review plans. The role of the IPEs in risk-informed regulation will be determined in the context of these documents, not NUREG-1560.

5. **Comment:** The report implies that until "quality" PRA requirements are fully met, PRAs cannot be used for any regulatory purposes. If that is the case, "as is" PRAs are inappropriate to support such areas as the Maintenance Rule and Technical Specification changes. Such an interpretation is counterproductive and is not supportive of the PRA Policy that looks to enhance use of PRA in regulation commensurate with the state-of-the-art technology. Recognized

weaknesses, and tools to deal with those weaknesses delineated in the Standard Review Plan makes the "as is" PRA applicable for a wide variety of applications while "quality" PRA requirements are phased in. Waiting until perfect "quality" of PRA is achieved before utilizing the results is impractical. It is expected that "quality" and "standardization " will evolve, not through a priori definition, but through frequent, repeated application and peer review of PRAs. (Reference: see Table C.1, #15, 17)

Response:

The comment is similar to comments received on Chapters 6 and 14 (refer to Comment #1 in Section C.6). These chapters and Chapter 8 have been significantly revised for the final report. It was not intended to imply that all the attributes in draft NUREG-1560 have to be met before a PRA can be used to support risk-informed regulatory applications.

6. **Comment:** The NUREG states the NRC staff plans to conduct follow-up activities to monitor implementation of the potential plant improvements identified by the IPEs. The improvements were identified as "potential improvements" which in most cases were identified as areas for further review. The NRC seems to be taking them as having been commitments. These improvements should not be treated as commitments unless the utility clearly identified them as commitments. (Reference: see Table C.1, #8)

   Response:

   The NRC recognizes that the potential improvements are not commitments in a regulatory sense. However, in many cases the improvements were credited in the IPE. Therefore, if the licensee uses the IPE in future submittals to the NRC, it is important for the NRC staff to know if the credited improvements have been made.

7. **Comment:** The use of NUREG-1560 for a variety of issues is discussed in Chapter 8. However, most of the discussions are actually related to the use of the IPEs to address these issues. NUREG-1560 should not be the source of information for applications as discussed in Chapter 8. The IPEs/PRAs are the primary source and should be used. (Reference: see Table C.1, #8)

   Response:

   NUREG-1560 summarizes a great deal of important safety information and provides a starting point for identifying and addressing a number of important safety issues. As such, it is an important document and staff resource. However, the staff recognizes that some of the information is out of date and that the individual submittals contain more information. For any particular issue, the staff will use the best available information, including any new submittals, recognizing that some of the new information may require additional review. NUREG-1560 also provides comparisons among the IPEs on selected issues, and this information is useful to the staff when evaluating the treatment of an issue by a particular plant.

C.11 Chapter 10: Background for Obtaining IPE Perspectives

Several comments were received concerning the accuracy of the information provided in this chapter. Corrections were made to the text where appropriate. No general comments were made concerning this chapter.

REFERENCES FOR APPENDIX C

C.1 Federal Register, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance, Summary Report, Draft," Vol. 61, No. 221, Page 58429, November 14, 1996.

Federal Register, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance, Volume 2, Parts 2-5, Draft," Vol. 61, No. 239, Page 65248, December 11, 1996.

C.2 Federal Register, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance, Volume 2, Parts 2-5, Draft," Vol. 61, No. 239, Page 65248, December 11, 1996.

C.3 NRC Memorandum (From Mary Drouin to M. Wayne Hodges), "Draft NUREG-1560 Public Workshop Summary Report," October 3, 1997.

C.4 ACRS meetings on IPE insights:

November 18, 1993	January 26, 1996 (subcommittee)
December 10, 1993	February 8, 1996
September 27, 1994	May 23, 1996
October 7, 1994	June 11, 1996 (subcommittee)
December 7, 1995	June 12, 1996

C.5 USNRC, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," Draft Regulatory Guide DG-1061, June 1997.

USNRC, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing," Draft Regulatory Guide DG-1062, June 1997.

USNRC, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance," Draft Regulatory Guide DG-1064, June 1997.

USNRC, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Draft Regulatory Guide DG-1065, June 1997.

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11. ABSTRACT *(200 words or less)*

This report provides perspectives gained by reviewing 75 Individual Plant Examination (IPE) submittals pertaining to 108 nuclear power plant units. IPEs are probabilistic analyses that estimate the core damage frequency (CDF) and containment performance for accidents initiated by internal events (including internal floods, but excluding internal fire). The U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research, reviewed the IPE submittals with the objective of gaining perspectives in three major areas: (1) improvements made to individual plants as a result of their IPEs and the collective results of the IPE program, (2) plant-specific design and operational features and modeling assumptions that significantly affect the estimates of CDF and containment performance, and (3) strengths and weaknesses of the models and methods used in the IPEs. These perspectives are gained by assessing the core damage and containment performance results, including overall CDF, accident sequences, dominant contributions to the design and operational characteristics of the various reactor and containment types, and by comparing the IPEs to probabilistic risk assessment characteristics. Methods, data, boundary conditions, and assumptions used in the IPEs are considered in understanding the difference and similarities observed among the various types of plants.

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