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

December 19, 2006

MEMORANDUM TO: Central Files (Attn: David Pinckney)

FROM: Jenny M. Gallo, Chief
Operations Support Branch, ACRS

SUBJECT: 538TH ACRS COMMITTEE FACA SUBMITTALS — 2006-10

In compliance with FACA § 10(b) and pursuant to the November 4, 1994, Memorandum of Understanding between the Advisory Committee on Reactor Safeguards/Advisory Committee on Nuclear Waste and the then Office of Information Resources Management, please ensure that the enclosed 538th ACRS Full Committee Meetings Notebooks are maintained in the NRC File Center until the ACRS ceases to exist.

Attachment:
Submittal 2006-10

R/3

DOCUMENTS DETERMINED TO BE RETAINED UNDER FACA REQUIREMENTS

SUBMITTAL NUMBER: 2006-10

I. DOCUMENTS FROM ACRS MEETINGS NUMBERS: 538TH

A. New Documents to be Placed in Central Files Only (Enclosed)

1. 538th ACRS Committee Meeting Notebook

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

November 20, 2006 (REVISED)

SCHEDULE AND OUTLINE FOR DISCUSSION
538th ACRS MEETING
DECEMBER 7-9, 2006

THURSDAY, DECEMBER 7, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)
   - 1.1) Opening statement
   - 1.2) Items of current interest

2) 8:35 - 10:30 A.M. Draft Final Regulatory Guide, DG-1145, "Combined License Applications for Nuclear Power Plants" (Open) (TSK/DCF)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff regarding Draft Final Regulatory Guide, DG-1145, "Combined License Applications for Nuclear Power Plants," and resolution of significant public comments.

   Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

10:30 - 10:45 A.M. ***BREAK***

3) 10:45 - 12:15 P.M. Draft Final Regulatory Guide, DG-1144, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors" (Open) (JSA/CGH/CS)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding Draft Final Regulatory Guide, DG-1144 and the resolution of public comments.

   Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

12:15 - 1:15 P.M. ***LUNCH***

4) 1:15 - 3:15 P.M. Proposed Revisions to Standard Review Plan Section 13.3, "Emergency Planning" (Open) (MLC/DAP/MB)
   - 4.1) Remarks by the Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the NRC staff regarding proposed revisions to Standard Review Plant Section 13.3, "Emergency Planning," and related matters.

   Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

3:15 - 3:30 P.M. *****BREAK*****

5) 3:30 - 5:30 P.M. State-of-the-Art Reactor Consequence Analysis Project (Open) (WJS/HPN)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding status of the staff's efforts associated with the state-of-the-art reactor consequence analysis project.

Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

5:30 - 5:45 P.M. *****BREAK*****

6) 5:45 - 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

6.1) Draft Final Regulatory Guide, DG-1145, "Combined License Applications for Nuclear Power Plants" (TSK/DCF)

6.2) Draft Final Regulatory Guide, DG-1144, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors" (JSA/CGH/CS)

6.3) Proposed Revisions to Standard Review Plan Section 13.3,"Emergency Planning" (MLC/DAP/MB)

6.4) State-of-the-Art Reactor Consequence Analysis Project (Tentative) (WJS/HPN)

6.5) Collaborative Research on Human Reliability Analysis Methods (GEA/EAT)

FRIDAY, DECEMBER 8, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

7) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)

8) 8:35 - 9:30 A.M. Proposed Revisions to Regulatory Guides and Standard Review Plan Sections in Support of New Reactor Licensing (Open) (OLM/DCF)

8.1) Remarks by the Subcommittee Chairman

8.2) Discussion of proposed revisions to Regulatory Guides and Standard Review Plan Sections that are being made in support of new reactor licensing.

9) 9:30 - 10:30 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/JTL/SD)

9.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.

9.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

10:30 - 10:45 A.M. ***BREAK***

10) 10:45 - 11:00 A.M. Reconciliation of ACRS Comments and Recommendations (Open) (GBW, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

11) 11:00 - 11:30 A.M. Election of ACRS Officers for CY 2007 (Open) (JTL/SD)
Election of Chairman and Vice-Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee.

11:30 - 1:00 P.M. ***LUNCH***

12) 1:00 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

12.1) Draft Final Regulatory Guide, DG-1145, "Combined License Applications for Nuclear Power Plants" (TSK/DCF)

12.2) Draft Final Regulatory Guide, DG-1144, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors" (JSA/CGH/CS)

12.3) Proposed Revisions to Standard Review Plan Section 13.3, "Emergency Planning" (MLC/DAP/MB)

12.4) State-of-the-Art Reactor Consequence Analysis Project (Tentative) (WJS/HPN)

12.5) Collaborative Research on Human Reliability Analysis Methods (GEA/EAT)

SATURDAY, DECEMBER 9, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

13)	8:30 - 12:00 Noon **(10:15-10:30 A.M. BREAK)**	Preparation of ACRS Reports (Open) Continue discussion of proposed ACRS reports listed under Item 12
14)	12:00 - 12:30 P.M.	Miscellaneous (Open) (GBW/JTL) Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for a specific item.. The remaining 50 percent of the time is reserved for discussion.
- Thirty-Five (35) hard copies and (1) electronic copy of the presentation materials should be provided to the ACRS.

November 16, 2006

OVERTIME SCHEDULE*
FOR THE 538th ACRS MEETING

THURSDAY, DECEMBER 7, 2006
Theron Brown
Sherry Meador
Jessie Delgado
Carol Brown

FRIDAY, DECEMBER 8, 2006
Theron Brown
Sherry Meador
Sonary Chey
Carol Brown

SATURDAY, DECEMBER 9, 2006
Theron Brown
Carol Brown
Barbara Jo White

* Any changes to above schedule should be checked with John T. Larkins.

November 17, 2006

COLOR CODE - 538th ACRS MEETING

Draft Final Regulatory Guide, DG-1145, "Combined License Applications for Nuclear Power Plants"	TSK/DCF	
Draft Final Regulatory Guide, DG-1144, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors"	JSA/CGH/CS	[illegible]
Proposed Revisions to Standard Review Plan Section 13.3, "Emergency Planning"	MLC/DAP/MB	**Green**
State-of-the-Art Reactor Consequence Analysis Project (Tentative)	WJS/HPN	**Violet**
Collaborative Research on Human Reliability Analysis Methods	GEA/EAT	Salmon

Letters may be added or deleted after Committee consideration.)

Tab 2

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
DRAFT REGULATORY GUIDE DG-1145
"COMBINED LICENSE APPLICATIONS FOR NUCLEAR POWER PLANTS (LWR EDITION"
NOVEMBER 30 , 2006
ROCKVILLE, MARYLAND

TABLE OF CONTENTS

Attachments:
1. S. Armijo's Specific Comments on DG-1145

Cognizant ACRS Member: Dr. Tom Kress

Cognizant ACRS Staff Engineer: David Fischer

538th ACRS Meeting
December 7, 2006
Rockville, MD

Draft Regulatory Guide DG-1145 (RG 1.206)

-PROPOSED AGENDA-

Cognizant Staff Engineer: David C. Fischer DCF@NRC.GOV (301) 415-6889

Topics		Presenters	Presentation Time
I	Opening Remarks	T. Kress, ACRS	8:35 am - 8:40 am
II	Staff Introductory Remarks	D. Matthews, NRR	8:40 am - 8:45 am
III	DG-1145 Overview - Purpose - Format and Structure - Developmental Basis - Status	E. Oesterle, NRR	8:45 am - 9:15 am
IV	PRA / RTNSS / RAP	D. Harrison, NRR P. Prescott, NRR	9:15 am - 9:45 am
V	TBD		9:45 am - 10:00 am
VI	Characterization of Public Comments	E. Oesterle, NRR	10:00 am - 10:15 am
VII	Industry Comments	Russell Bell, NEI	10:15 am - 10:25 am
VIII	Summary	T. Kress	10:25 am - 10:30 am
IX	BREAK		10:30 am - 10:45 am

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.
- 35 copies of the presentation materials to be provided to the Subcommittee.

538TH MEETING OF THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
DECEMBER 7, 2006
ROCKVILLE, MARYLAND

DRAFT REGULATORY GUIDE DG-1145 (RG 1.206)

- STATUS REPORT -

The purpose of Draft Regulatory Guide DG-1145, "Combined License Applications for Nuclear Power Plants (LWR Edition)," is to provide guidance regarding the information to be submitted in a Combined License (COL) application for a nuclear power plant. As such, this guide is intended to address many, albeit not all, of the application options allowed by the Code of Federal Regulations, Title 10, Part 52. Although a COL applicant is not required to conform to this guidance, its use will facilitate both the applicant's preparation of a COL application and timely review of the application by the NRC staff.

The COL application is comprised of the various application items listed below:

Final Safety Analysis Report
Inspections, Tests, Analyses, and Acceptance Criteria
Probabilistic Risk Assessment
Environmental Report
Security Plan
General and Financial
Quality Assurance Program Description

This draft Regulatory Guide was made publicly available on September 1, 2006, on the NRC website and the 45 day public comment period officially began on September 6, 2006, upon posting in the Federal Register. The public comment period closed on October 23, 2006.

By and large, DG-1145 contains guidance on what to include in a COL application. It generally does not contain the acceptance criteria or review procedure for evaluating the information provided in the COL application. The acceptance criteria and review procedures would be contained in the related Regulatory Guides and Standard Review Plan Sections. The Committee underwent a separate effort to review the "high-priority" regulatory guides and standard review plan (SRP) sections being developed or revised in support of new reactor licensing. The staff has made, and will continue to make, every effort to ensure that the scope and level of detail of the information to be provided in the COL application (i.e., as described in DG-1145) is consistent with the guidance and acceptance criteria provided in the revised (or new) regulatory guides and SRP sections.

In the public meetings on September 22 and October 3, 2006, NEI discussed three steps for continuing the public engagement on DG-1145.

1. Hold public meetings in December 2006 and, if necessary, in January 2007, to discuss the disposition of the stakeholder comments and open items.
2. Post a draft final version that reflects the reconciliation of public comments on the NRC website as soon as possible.

3. Hold a final meeting following issuance of the final regulatory guide to explain changes made in finalizing the document, including those changes made to conform the guidance to the final Part 52 rule.

NEI said that the industry is committed to continue to work with the NRC staff on the development of guidance for COL applicants. This will assure clarity and a common understanding of the key elements of the regulatory infrastructure, including DG-1145, the Standard Review Plan, Part 52 and related NRC regulations. NEI said that this is essential for assuring the development of quality combined license applications and for assuring NRC reviews are conducted in the most effective and efficient manner.

STRUCTURE OF DG-1145

The regulatory positions presented in Section C of this guide are divided into four parts.

Part I, Col Applicants Who Are Not Referencing Certified Designs

Part I addresses the information requirements specified in 10 CFR 52.79, "Contents of Applications; Technical Information in Final Safety Analysis Report." Part I is intended to provide COL applicants with guidance regarding the information that the staff needs to resolve all safety issues related to the proposed combined license. Moreover, Part I is intended to be used by COL applicants who are ***not*** referencing certified designs. Part I includes 19 sections. Section C.I.1 provides broad generic guidance, although COL applicants have the option not to maintain some of this information in Chapter 1 of the final safety analysis report (FSAR). Sections C.I.2–C.I.17 are based on the existing guidance provided in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," although the NRC staff has updated the guidance in those sections to reflect the current information requirements for COL applications. By contrast, Sections C.I.18–C.I.19 present information requirements that are not addressed in Regulatory Guide 1.70. In addition, the reader should note that Sections C.I.2–C.I.19 correspond to Chapters 2–19 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants." The level of information needed for those sections depends on the complexity of the topic.

Part II, COL Applicant Referencing a Custom Design

Part II of Section C addresses the information requirements specified in 10 CFR 52.80, "Contents of Applications; Additional Technical Information." In particular, these information requirements include the probabilistic risk assessment (PRA); inspections, tests, analyses, and acceptance criteria (ITAAC); and the environmental report. Use of the guidance in Part II assumes that a COL applicant is referencing a custom design. Together, Parts I and II are intended to represent the bulk of the technical information that an applicant should include in a COL application.

Part III, COL Applicants Who Reference Either A Certified Design or Both a Certified Design and an Early Site Permit (ESP)

Part III of Section C is intended to be used by COL applicants who reference either a certified design or both a certified design and an early site permit (ESP). Part III includes seven sections.

Section C.III.1 is intended to address the topics that the NRC staff will review in a COL application that references a certified design. By contrast, Section C.III.2 addresses the remaining review topics for applications that reference both a certified design and an ESP. The guidance provided in both of these sections was derived from information presented in Part I of this guide. Section C.III.3 addresses the finality of an environmental impact statement associated with an ESP. Section C.III.4 provides generic guidance on addressing COL action/information items in COL applications. Section C.III.5 provides recommendations for COL applicants who reference certified designs that include design acceptance criteria (DACs). Section C.III.6 provides recommendations for coordinating the submittal of COL applications with design certifications and/or ESP applications that are under NRC review at the time the COL application is submitted. Finally, Section C.III.7 provides a process for developing the additional ITAAC necessary for applications that reference a certified design.

Part IV, Miscellaneous Topics Of interest to Col Applicants

Part IV of Section C includes 12 sections that address a series of miscellaneous topics of interest to COL applicants. Section C.IV.1 includes the checklist that the NRC will use to perform its acceptance review of a COL application. Section C.IV.2 provides guidance and recommendations for the format of a COL application, with a particular focus on those that applicants submit electronically. .Section C.IV.3 provides a general description of the change processes associated with custom COL applications and those that reference a certified design and/or an ESP. Section C.IV.4 provides guidance for use in implementing SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria." Section C.IV.5 provides submittal guidance for the general and financial information that a COL application is required to include. Section C.IV.6 provides guidance regarding information to be included in the site redress plan and requests for limited work authorizations. Section C.IV.7 discusses pre-application activities that the NRC staff and the prospective applicant should perform before an application is submitted. Section C.IV.8 provides information on dealing with generic issues. Section C.IV.9 is reserved for future use. Section C.IV.10 provides guidance on handling the regulatory treatment of non-safety systems. Section C.IV.11 is reserve for future use. Finally, Section C.IV.12 discusses the applicability of industry guidance. Appendix A provides the questions and comments received during the public workshops on DG-1145 and the proposed staff responses.

Appendix I (or possibly A), Responses to Public Comments on DG-1145

The comments and draft responses in the Appendix are organized by their corresponding section in DG-1145 (e.g., C.I.1, C.I. 2, C.I.3, ... , C.II.1, CII.2, C.II.3, ... , C.III.1, CIII. 2, ...).

<u>**NEI COMMENTS ON DG-1145**</u>

The industry recognizes that the September 2006 draft is still a work-in-progress and the open issues are numerous, as reflected in our detailed comments that are described in the Enclosure. As a result, it is important that public interactions continue so that a common understanding is established between the NRC staff and the industry on what constitutes a complete, practical and quality combined license application.

Our main comments are:

1) The guidance seeks information for combined license (COL) applications that will not be

available at time of a COL application submittal. In each case, we believe that alternative information may be provided in the COL application to support NRC safety reviews or the information sought may be verified by the staff as part of design implementation inspections after COL issuance.

2) Part III of the guidance, which assumes a design certification is referenced, seeks COL application information on matters that have been resolved during the design certification proceedings. This is contrary to the Part 52 principle of design certification finality, which provides that no additional detail is required in COL applications on the approved standard design. Examples of this are identified in Comments C.III.1.47-48 in the enclosure.

3) The guidance seeks similar information about off-site AC power sources for both evolutionary and "passive" plant designs. Passive plants do not rely on off-site AC power for any safety function, as a result the information required about off-site power sources should be much less. Comment C.1.8.5.

4) Several sections of draft DG-1145 refer to corresponding Standard Review Plan (SRP) Sections that are currently being drafted and not available for review at this time. As such, the industry is unable to provide meaningful comment at this stage. This comment emphasizes the importance of holding additional public meetings on DG-1145 and specific SRP sections.

5) Plant-Specific PRA:

a. *Large Release Frequency (LRF):* The guidance introduces a new PRA metric, LRF for evaluating changes to the licensing basis during operations. The development of Reg. Guide 1.174 and the ASME Standard RA-Sb-2005, PRA *Internal Events,* which will be endorsed in Reg. Guide 1.200, has taken many years. In that period of development, the use of LRF as a metric for operational decision-making was evaluated. It was rejected in favor of core damage frequency and large early release frequency. To propose the LRF metric so shortly after it was rejected for use in operational assessments is disconcerting.

A more precise and consistent definition of LRF would have to be developed for use in an operational setting compared with the definitions that were developed for design certifications. This would require substantial interaction with the PRA technical community before a common understanding could be reached on such a definition and how it would be applied. This would introduce uncertainty at a critical time in the new licensing process as applicants start on the final drafts of their applications that will be submitted next year. The guidance should use the same metrics that are used for existing plants for evaluating changes to the licensing basis in the operational phase: Large Early Release Frequency, which corresponds to early health effects, and Core Damage Frequency.

b. *Conditional Containment Failure Probability (CCFP):* The draft guidance proposes the CCFP of 0.1, given a core melt. For advanced designs, whose calculated internal event core damage frequency is approximately 10^{-7} /year, the CCFP would translate into a containment failure frequency of approximately 10^{-8} /year. It is impractical and unreasonable to attempt to design a containment structure to withstand naturally occurring ultra-low frequency events of this magnitude, for example a one in a 100 million-year earthquake. Hence, as interpreted by the industry, the proposed CCFP could not be met. There is a need for further industry-NRC interaction on developing a practical containment performance metric that could be used in operational licensing evaluations for designs that have very low core-damage frequency.

c. *COL PRA Information:* The guidance should clarify that no additional plant-specific PRA information is required to be included in the COL application where the design certification PRA bounds the site- and plant-specific parameters.

6) ITAAC:

a. Section C.11.2 contains numerous examples of incorrect criteria for establishing ITAAC. The proposed criteria do not meet the well established criteria for ITAAC described in SRP 14.3 and in Generic Design Control Documents, Section 14.3. ITAAC are established to verify top-level (Tier 1) design descriptions and performance standards. Examples of this problem are in Section C.II.2.2.5 and Section C.II.2, Attachment A, on ITAAC for Instrumentation and Controls, which-call for ITAAC on "cabinet layout and wiring" and other second-tier design information.

b. The guidance should not call for additional ITAAC at the COL stage on matters that were resolved through a referenced design certification. This would be contrary to the Part 52 design finality principle. At the time of COL, ITAAC are developed for emergency planning and the site-specific design, including physical security features, as appropriate, in accordance with the criteria in SRP Section 14.3. Moreover, whether a design certification is referenced or not, the lack of complete detailed design information in a COL application is not a basis for requiring ITAAC. Applications will contain sufficient information to support required NRC safety findings, recognizing the NRC staff will have an opportunity later to verify the design implementation through the Construction Inspection Program.

7) DG-1145 contains placeholders in Sections C.II.3 and C.III.3 for guidance on COL application environmental reports. Attaining a common understanding on environmental matters, including how Early Site Permit environmental finality will be assessed at the time of a COL, is critical for assuring an effective and efficient licensing proceeding. Also, the benefit and effectiveness of the Early Site Permit subpart in Part 52 will hinge on how this section will be interpreted by the industry and NRC staff. The importance of these two sections underscores the importance of having additional public meetings.

8) NRC rules and DG-1145 guidance for addressing Regulatory Guides, SRPs and operating experience require an applicant to address the guidance in effect six months before docket date. However, "six months before docket date" is not a fixed date known by either the NRC or applicant. The NRC staff has stated that it has been and will continue to be the NRC practice to implement this requirement as "six months before application date." DG-1145 and existing regulations should be modified to reflect NRC practice and intent going forward. (Comments C.I.1.10 and C.III.1.9)

9) Appendix I documents NRC responses to numerous comments raised in connection with the DG-1145 workshops. In many instances, the NRC staff agreed with the comment but

declined to change or modify the guidance. For example, the NRC staff response to workshop Comment C.I.13.1.2.1-1 agreed that a high level organization chart is sufficient to provide in Section 13.1 of the FSAR. However, the staff declined to modify Section C.I.13.1, which currently seeks a more detailed organization chart than COL applicants will have developed at the time of application submittal.

It is vital that DG-1145 document the understandings reached during the workshops and public comment process. Failure to do so will cause misinterpretations in the future, unnecessarily prolonging licensing proceedings.

The industry comments in Enclosure 1 to NEI's October 20, 2006, letter are organized by DG-1145 section and prioritized high, medium and low (1, 2 or 3). The comments fall into four general categories ("Basis Codes").

1) The guidance does not conform with the regulations
2) The guidance seeks information that will not be available at the time of COL application submittal and is not necessary to support required reasonable assurance findings
3) The guidance is not consistent with other NRC guidance
4) Clarification is needed

Where a comment is a follow-up to an NRC response provided in Appendix I of DG-1145, NEI included the NRC's Appendix I comment number for reference.

Attachment 1 to the Enclosure provides mark-ups of specific DG-1145 sections consistent with the comments in Enclosure 1. The specific sections are: C.III.8, Electric Power; C.I.11, Radioactive Waste Management; C.I.17.6, Description of Applicant's Program for Implementation of 10 CFR 50.65, the Maintenance Rule; and C.I.18, Human Factors Engineering.

ACRS MEMBER COMMENTS

Jack Sieber 11/07/2006 3:44 PM

As you know, these sections are difficult to review, since we do not have a library of applicable codes and standards at home, and I need to use the NRC web site (which doesn't always work properly) to view the referenced Regulatory Guides. However, I am familiar with most of the applicable codes and standards which are referenced in the two sections assigned to me.

What I notice most are the things that seem to be missing. For example, in the reactor coolant system section, I did not see any requirement for the applicant to supply drawings of hangers, supports (both pipe and component supports) and restraints (whip (if any) and seismic restraints). Also, when describing the materials of the system, I think that iso drawings are very helpful, if they are annotated with base and weld metal compositions.

Also, besides knowing the chemistry, it is good to have a record of pre and post fabrication and weld heat treatments. Also, a description of any repairs made (and what sections of the ASME Code (or Code Cases) allow the repair to be made). There are other issues that should be elaborated on in this section, related to its fabrication, erection, testing and qualification of the piping, components and supports.

In the area of Electric Power Systems, I would like to see the Applicant describe its plant electrical grounding system. I would like to see some typical three wire diagrams along with a full set of single line drawings which show the protection components.

I also believe that the analysis required for grid stability calcs does not fully bound all possible vulnerabilities. Perhaps what is asked for is the best that one can do, since we do not describe a real design basis as to what we expect the electric power grid to be able to do. Since many applicants do not have control of the grid to which their plant is connected. Again, perhaps this is the best that can be done.

Also, there is a standard referenced for coordinated electrical protection, but no description of the electrical protection scheme is requested. Since electrical protection is well understood, perhaps no further information is needed - except for single line drawings.

I did not find any reference to fire protection except a reference as to the separation criteria for electrical systems. Where does a description of the fire protection program come in?

I also reviewed the I & C section. I have some comments, but since this section was not assigned to me, I will not comment unless some missing piece is evident. Otherwise, the I & C section looks OK to me, without breaking new ground. The description matches the rules in place today.

I wonder if a one day meeting will be enough to review this document? I think that there is a lot of information contained in it and that the review is complex, since it has so many references. Does NRR plan to discuss each section individually? If so, in what detail? If not, are we just to show up with our list of questions?

I think that it would be a good idea for us to know exactly what NRR plans to say so that we can prepare properly.

Said Abdel-Khalik 11/12/2006 6:02 PM

I have reviewed Chapters 7 (Instrumentation & Controls) and Chapter 10 (Steam and Power Conversion System) of DG-1145. The following comments are offered:

I. Chapter 7 -- Instrumentation & Controls:

My questions pertain to items 6 and 7 of Appendix C.I.7-A (Digital Instrumentation and Control Systems Application Guidance).

Item 6 deals with life cycle process requirements; it specifies that " ... The sample size should be such that the staff can conclude with at least 95% assurance that the quality of the design has been validated."

Item 7 deals with software life cycle process design outputs; it specifies that ".... A statistically valid sample of software design outputs should be provided to confirm with at least 95% assurance that they address that they address the functional requirements and .... "

The questions in both cases deal with the specified 95% confidence level. Why was it selected? and is it adequate for all systems including those important to safety?

II. Chapter 10 (Steam & Power Conversion System)

Section C.I.10.3.5 deals with Water Chemistry (PWR only). There is no section dealing with BWR water chemistry. Will that information be provided in a different Chapter?

Bill Shack 11/12/2006 11:31 PM

Agenda appears OK. It is hard to know without a better overview of the whole document than I have at the moment. Others will have similar problems. The information on pipe whip restraints, etc. that Jack was looking for in the reactor coolant section is in Chapter 3.

A few preliminary comments on Chapter 3, which seems to me quite comprehensive and has been updated to reflect recent experience.

The leak-before-break discussion is pretty good in terms of all the items that should be considered, but why after 20 years or so is there no RG on LBB. One appeared imminent about 5 years ago, but nothing appears to be even an official draft. I expect all applicants to try to take advantage of LBB as much as possible. Maybe licensees and the staff have enough experience after all the submittals that have been made and reviewed, but then it would possible to formalize it in a RG.

Discussion on flow induced loadings now includes acoustic modes as well as flow induced vibrations and steam dryers are highlighted.

Why no reference to the RG (and Draft RG) on tornado winds in 3.3?

Discussion of aircraft hazards is interesting but only addresses accidents:

"The aircraft hazard analysis should provide an estimate of the total aircraft hazard probability per year. Aircraft accidents that could lead to radiological consequences in excess of the exposure guidelines of 10 CFR 50.34(a)(1) with a probability of occurrence greater than an order-of-magnitude of 10^{-7} per year should be considered in the design of the plant. Provide and justify the aircraft selected as the design-basis impact event, including its dimensions, mass (including variations along the length of the aircraft), energy, velocity, trajectory, and energy density."

Tom Kress 11/13/2006 1:08 PM

I have reviewed my chapters of DG-1145 which are:

1. Introduction & General Description of Plant
14. Initial Test Program and ITAAC -- Design Certification
20. Generic Issues.

Believe it or not, I have no comments on these chapters. If the rest of the reviewers have the same reaction, it is going to be a dull meeting. I presume the process will be that the staff will give a very brief statement of what is in a chapter and then ask for any comments. I liked what Jack Sieber did in looking for what was missing rather than what was there. Perhaps I could persuade him and Otto to look at Chapter 14 which, by the way, is called Verification Program in the DG. They both know a hell of a lot more about that subject than I do which is very little.

My only real issue with the DG is that there should be a requirement and guidance on having to do a site specific Level-3 risk assessment. This may very well be a part of the Environmental Impact Statement but I think it should be explicitly stated in this RG.

Michael Corradini 11/13/2006 1:42 PM

I have looked over my two sections (assignments are section 6 and 19.2) and my reactions are similar for section 6.

In the area of Engr. Safety Features, the guidance is very specific and complete and I really do not have any major issues. I must admit that I am not experienced on what had been specified in past documents, but it is my experience that this guidance is much more complete and is a result of experience from submittals.

My observations for section 19.2 is the exact opposite - there is nothing really there in any substantive way. I looked at 19.1 and I think that the two are a package and I would wonder what GA thinks? I do not see spending the whole day on this if all of our sections have these binary results.

Sam Armijo 11/13/2006 2:48 PM

I have reviewed Chapter 4 (Reactor) and have found it to be pretty complete. I have included my comments in the attachment (Attachment 1). I was not impressed with the level of information requested for materials in this chapter, but found a more complete list of requested information in Chapter 5 (Reactor Coolant and Connecting Systems) which Jack is reviewing. I will
take a closer look at this and send you my comments later.

It seems to me that Design Guide should have a special chapter that asks the applicant to identify all the materials degradation mechanisms that have plagued the industry and to describe what materials selections, fabrication steps, mechanical design, and water chemistry specifications they have put it place to prevent failures in new reactors. For example the chapter would list the known failure mechanisms each reactor type (for example, IGSCC, IASCC, FAC, PWSCC, Thermal Fatigue, denting, vessel embrittlement, etc) and describe why the new materials would not be subject to each mechanism. I think it would help the designers focus their materials efforts, demonstrate that they have put these problems to bed, and help the staff determine if the designs are complete and adequate. Right now it seems that material issues are sprinkled all over the DG.

The industry has spent billions fixing materials failure and degradation issues, and the DG should ask for a comprehensive treatment of these issues.

D. C. Fischer, ACRS Staff

Section C.II.1, "Probabilistic Risk Assessment (PRA)" identifies 8 objectives that the COL applicant's risk evaluation are supposed to meet. It is not clear to me how practical this guidance will be to COL applicants, inasmuch as neither the ASME PRA Standard (ASME RA-S-2002) nor Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" are not constructed in these terms. In addition, while DG-1145 provides PRA-related guidance on the content of a COL application, there does not appear to be review procedures and acceptance criteria (e.g., SRP guidance) for determining whether an applicant's submittal meets these objectives. Finally, the second objective, "Determine how the risk associated with design relates to the Commissions goals of less than 1 E-4/yr for core damage frequency (CDF) and less than 1 E-6/yr for large release frequency (LRF)," introduces what could be a technically challenging and potentially problematic metric (i.e., LRF). From an editorial perspective, this section often refers to "the COL applicant" when it should be referring to the "COL holder" (e.g., in discussing PRA updates).

Section C. IV.10, "Regulatory Treatment of Non-Safety Systems," provides guidance to COL applicants, that do not reference a certified design, on the process that should be used to determine which non-safety SSCs should receive regulatory treatment. However, there does not appear to be review procedures and acceptance criteria (e.g., SRP guidance) for assessing an applicant's RTNSS process. In addition, there does not appear to be any guidance for assessing the acceptability of an applicant's proposed reliability/availability missions, treatment, and regulatory oversight that should be applied to such equipment.

EXPECTED COMMITTEE'S ACTION

The Full Committee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, and may prepare a letter to the EDO on DG-1145.

<u>References</u>

1. Memorandum dated September 1, 2006, from David B. Matthews, Director, Division of New Reactor Licensing, NRR to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Draft Regulatory Guide DG-1145 "Combined License Applications for Nuclear Power Plants (LWR Edition)" (ML062440120)
2. Letter dated October 20, 2006, from Adrian Heymer, Nuclear Energy Institute to Rules and Directives Branch, Office of Administration, NRC, Subject: Notice of Availability Draft Regulatory Guide DG-1145 "Combined License Applications for Nuclear Power Plants (LWR Edition)." ("DG-1145") 71 Fed. Reg. 52,826 (Sept. 7, 2006). (ML063000204)

November 13, 2006

S. Armijo's Specific Comments on DG-1145

C.I.4.2, Fuel System Design

Should include information related to effects of oxidation and hydriding on the mechanical properties of the fuel cladding.

C.I.4.2.3, Design Evaluation

(1) Cladding

(e) With regard to stress-accelerated corrosion:

This is a strange term for fuel cladding. Should use stress corrosion cracking due to pellet clad interaction (PCI)

(g) With regard to material wastage due to mass transfer:

I never heard of a mass transfer phenomenon. Are the authors talking about accelerated cladding corrosion due to excessive crud deposition and subsequent burnout? If so should ask for info on crud-related failure mechanisms.

Also discuss the following phenomenological models:

• fuel and cladding temperature distribution

Probably mean radial power distribution.

(1) Fuel system damage criteria for all known mechanisms:

Should insert "failure or performance limiting mechanisms".

(b) commutative number of strain fatigue cycles

Should be cumulative number of cycles.

Describe the processes, inspections, and tests used to ensure that austenitic stainless steel components are free from increased susceptibility to intergranular stress-corrosion cracking caused by sensitization. If special processing or fabrication methods subject the materials to temperatures between 800–1,500°F (427–816°C), or involve slow cooling from temperatures over 1500°F (816°C), describe the processing or fabrication methods and provide justification to show that such treatment will not cause susceptibility to intergranular stress-corrosion cracking. Indicate the degree of conformance to the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," as well as Position C.5 of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of

Water-Cooled Nuclear Power Plants," as it relates to controls for abrasive steel surfaces. Provide justification for any deviations from these recommendations.

This paragraph should be changed to read " Describe the processes, inspections, and tests used to ensure that austenitic stainless steels are highly resistant to intergranular stress corrosion cracking, and that fabrication or special processing methods such as heat treatment, cold work, welding or post-weld griding do not create IGSCC susceptibility in the reactor coolant environment."

A similar statement should be written to address PWSCC in PWR coolants.

Tab 3

**NOTEBOOK MATERIAL
FOR THE ACRS FULL COMMITTEE MEETING
RELATED TO THE REVIEW OF REGULATORY GUIDE 1.207 (DG-1144), "GUIDELINES FOR EVALUATING FATIGUE ANALYSES INCORPORATING THE LIFE REDUCTION OF METAL COMPONENTS DUE TO THE EFFECTS OF THE LIGHT-WATER REACTOR ENVIRONMENT FOR NEW REACTORS"**

Ref: The following documents were sent previously:

1) NUREG/CR-6909 Rev. 1, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials"

2) Staff Response to Public Comments on DG-1144 and Draft NUREG/CR-6909

3) Redline-strikeout comparison between DG-1144 and draft final Regulatory Guide 1.207

Cognizant ACRS Member: Joe Armijo
Cognizant ACRS Staff Engineers: Gary Hammer

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
REVIEW OF REGULATORY GUIDE 1.207 (DG-1144)
GUIDELINES FOR EVALUATING FATIGUE ANALYSES INCORPORATING THE LIFE REDUCTION OF METAL COMPONENTS DUE TO THE EFFECTS OF THE LIGHT-WATER REACTOR ENVIRONMENT FOR NEW REACTORS
December 7, 2006
ROCKVILLE, MARYLAND

-PROPOSED SCHEDULE-

Cognizant Staff Engineer: Charles G. Hammer, cgh@nrc.gov (301) 415-7363

	Topics	Presenters	Time
I.	**Opening Remarks**	S. Armijo, ACRS	10:45 - 10:50 am
II.	**Overview of Regulatory Guide 1.207 (DG-1144)**	H. Gonzalez, RES	10:50 - 11:20 am
III.	**Discussion of technical basis for RG 1.207 and NUREG/CR-6909**	H. Gonzalez, RES O. Chopra, Argonne National Laboratory	11:20 - 12:00 pm
IV.	**Committee Discussion**	S. Armijo, ACRS	12:00 - 12:15 pm

Note

- Presentation time should not exceed 50 percent of the total time allocated for specific items. The remaining 50 percent of the time is reserved for discussion.
- 35 copies of the presentation materials to be provided to the Committee.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
REVIEW OF REGULATORY GUIDE 1.207 (DG-1144)
GUIDELINES FOR EVALUATING FATIGUE ANALYSES INCORPORATING THE LIFE
REDUCTION OF METAL COMPONENTS DUE TO THE EFFECTS OF
THE LIGHT-WATER REACTOR ENVIRONMENT FOR NEW REACTORS
December 6, 2006
ROCKVILLE, MARYLAND

- STATUS REPORT -

PURPOSE:

The purpose of this session is to review the Regulatory Guide 1.207 (DG-1144) being developed by the Office of Research (RES) and their contractor, Argonne National Laboratory (ANL), to address the effects of the reactor coolant environment on the fatigue life of metal reactor coolant pressure boundary materials. Earlier versions of DG-1144 included guidance for carbon steel, low-alloy steel, and austenitic steel materials, and the current version of RG 1.207, also includes guidance for nickel-chromium-iron (Ni-Cr-Fe) alloy materials. The committee will hear presentations by and hold discussions with representatives of RES, ANL, and the American Society of Mechanical Engineers (ASME).

During the 532nd ACRS meeting in May 2006, the committee had no objection to the staff's proposal to issue DG-1144 for public comment and wished the opportunity to review the draft final version after reconciliation of public comments. In the interim, the RES staff has issued draft DG-1144 and has received several comments from the public. The staff has also prepared draft responses to these comments.

BACKGROUND:

In Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50), " Domestic Licensing of Production and Utilization Facilities," General Design Criterion (GDC) 1, "Quality Standards and Records," requires, in part, that structures, systems, and components that are important to safety must be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function performed. In addition, GDC 30, "Quality of Reactor Coolant Pressure Boundary," requires, in part, that components that are part of the reactor coolant pressure boundary must be designed, fabricated, erected, and tested to the highest practical quality standards.

Augmenting those design criteria, 10 CFR 50.55a, " Codes and Standards," endorses the ASME Boiler and Pressure Vessel Code for design of safety-related systems and components. In particular, Section 50.55a(c), " Reactor Coolant Pressure Boundary," requires, in part, that components of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III, "Rules for Construction of Nuclear Power Plant Components," of the ASME Boiler and Pressure Vessel Code. Specifically, those Class 1 requirements contain provisions, including fatigue design curves, for determining a component's suitability for cyclic service. These fatigue design curves are based on strain-controlled tests performed on small polished specimens, at room temperature, in air environments. Thus, these curves do not address the impact of the reactor coolant system environment.

RG 1.207 provides guidance for use in determining the acceptable fatigue life of ASME pressure boundary components, with consideration of the light-water reactor (LWR) environment. In so doing, this guide describes a methodology that the NRC staff considers acceptable to support reviews of applications that the agency expects to receive for new nuclear reactor construction permits or operating licenses under 10 CFR Part 50, design certifications under 10 CFR Part 52, and combined licenses under 10 CFR Part 52 that do not reference a standard design. Because of significant conservatism in quantifying other plant-related variables (such as cyclic behavior, including stress and loading rates) involved in cumulative fatigue life calculations, the design of the current fleet of reactors is satisfactory, and the plants are safe to operate.

DISCUSSION:

The ASME Section III design curves, developed in the late 1960s and early 1970s, are based on tests conducted in laboratory air environments at ambient temperatures. The original code developers applied margins of 2 on strain and 20 on cyclic life to account for variations in materials, surface finish, data scatter, and environmental effects (including temperature differences between specimen test conditions and reactor operating experience). However, the developers lacked sufficient data to explicitly evaluate and account for the degradation attributable to exposure to aqueous coolants. More recent fatigue test data from the United States, Japan, and elsewhere show that the LWR environment can have a significant impact on the fatigue life of carbon and low-alloy steels, as well as austenitic stainless steel and nickel alloy materials.

Two distinct methods can be used to incorporate LWR environmental effects into the fatigue analysis of ASME Class 1 components. The first method involves developing new fatigue curves that are applicable to LWR environments. Given that the fatigue life of ASME Class 1 components in LWR environments is a function of several parameters, this method would necessitate developing several fatigue curves to address potential parameter variations. An alternative would be to develop a single *bounding* fatigue curve, which may be overly conservative for most applications. The second method involves using an environmental correction factor (Fen) to account for LWR environments by correcting the fatigue usage calculated with the ASME "air" curves. This method affords the designer greater flexibility to calculate the appropriate impacts for specific environmental parameters. In addition, applicants have already used this method in their license renewal applications.

The NRC staff has selected the Fen method, as described in NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials." In particular, Appendix A to that report describes a methodology that the staff considers acceptable to incorporate the effects of reactor coolant environments on fatigue usage factor evaluations of metal components. In addition, NUREG/CR-6909 provides a comprehensive review of, and technical basis for, the methodology proposed in RG 1.207, including analysis of each parameter affecting the fatigue evaluations. In developing the underlying Argonne National Laboratory (ANL) models, the researchers analyzed existing data to predict fatigue lives as a function of temperature, strain rate, dissolved oxygen level in water, and sulfur content of the steel. The resultant method postulates a strain threshold, below which environmental effects on fatigue life do not occur. By definition, Fen is the ratio of fatigue life of the component material in a room temperature air environment to its fatigue life in LWR coolant at operating temperature. To incorporate environmental effects into the fatigue evaluation, the fatigue usage is calculated using ASME Section III Code provisions, and the fatigue design curve is multiplied by the correction factor.

A second concern regarding the ASME fatigue design curves involves nonconservatism of the current ASME stainless steel air design curve. More recent evaluations of stainless steel and nickel alloy test data indicate that the ASME curve is inconsistent with the appropriate test materials and conduct of the fatigue test. Consequently, through RG 1.207, the NRC staff endorses a new stainless steel air design curve. Section 5.1.8 of NUREG/CR-6909 provides a comprehensive review of, and technical basis for, that new design curve. The Fen values defined for stainless steel in NUREG/CR-6909 should be used in conjunction with the new stainless steel air design curve when evaluating the fatigue usage of ASME Class 1 components.

In addition, the staff evaluated the incorporation of the Fen approach methodology in fatigue analyses for Ni-Cr-Fe alloys (e.g., Alloy 600 and 690) and welds. Section 6 of NUREG/CR-6909 discusses the technical basis for incorporating the environmental effects on nickel alloys and welds. In summary, fatigue evaluations for Ni-Cr-Fe alloys are based on the fatigue design curve for austenitic stainless steels. However, the existing fatigue data for Ni-Cr-Fe alloys and their welds are not consistent with the current ASME Code fatigue design curve for austenitic stainless steels. The data are either comparable or slightly conservative with the updated ANL model for austenitic stainless steels. Thus, the new fatigue design curve proposed for austenitic stainless steels adequately represents the fatigue behavior of Ni-Cr-Fe alloys and their welds. Therefore, the new design curve for austenitic stainless steels may also be used for Ni-Cr-Fe alloys and their welds. The staff finds it acceptable to use the new austenitic stainless steels air design curve in Ni-Cr-Fe alloys environmental fatigue evaluations. Consequently, Section 6 of NUREG/CR-6909 presents the respective Fen equations to be used for Ni-Cr-Fe alloys and their welds.

Section 7 of NUREG/CR-6909 evaluates the ASME design curve margins. In conducting that evaluation, researchers reviewed data available in the literature to assess the subfactors (excluding environment) necessary to account for the effects of various uncertainties and differences between actual components and laboratory test specimens. The researchers also performed statistical analyses using Monte Carlo simulations to develop fatigue design curves, using the " 95/95 criterion". In other words, the curves should provide 95% confidence that 95% of the population will have a greater fatigue life than predicted by the design curves. The NRC staff deems this criterion acceptable because the fatigue design curves are based on crack initiation, rather than component failure, and therefore, additional margin exists between crack initiation and actual component failure.

The results of the Monte Carlo simulations indicate that for both carbon steels, low-alloy steels, and austenitic stainless steels, the current ASME Code procedure of adjusting the mean test data by a factor of 20 for life is conservative compared to the 95/95 criterion. The results also indicate that a minimum factor of 12 for cyclic life of these materials will satisfy the 95/95 criterion. Figures 9, 10, and 37 of NUREG/CR-6909 present the resultant new air design curves, using margins of 12 for life and 2 for stress, for carbon steel, low -alloy steel, austenitic stainless steel, respectively. RG 1.207 uses these new air design curves, thus, an applicant that chooses to adopt the guidance procedure to determine the fatigue life of stainless steels, these air design curves should be used. However, the existing ASME air design curves for carbon and low-alloy steels may also be used with the procedure in this guide to determine the fatigue life of those materials, since their use will yield conservative results.

The NRC staff reviewed and found acceptable several methods for calculating Fen. Only the types of stress cycles or load set pairs that exceed strain threshold criteria for carbon steels, low-alloy steels, and austenitic stainless steels need to be considered for Fen calculations. The evaluation options depend on the complexity of the analyzed transient condition and the detail of the evaluation. For example, in an evaluation in which the results of detailed transient analyses are available to determine the necessary parameters (strain rate, temperature, and others), the "modified rate approach" (presented and referenced in Section 4.2.14 of NUREG/CR-6909) is an acceptable methodology for determining the Fen values. This methodology involves a strain-based integral for evaluating conditions for which temperature and strain rate change, resulting in variation of Fen over time. This detailed approach calculates the Fen values based on the strain history for each load set in the fatigue analysis evaluation, considering the effects of strain rate and temperature variations for each incremental segment in the strain history. Such results may be used to reduce the conservatism in the calculated Fen values. For a simplified calculation yielding a more conservative result for a complex or poorly defined set of transients, the temperature is equal to the average temperature in the transient or segment. The calculated Fen values are then used to incorporate environmental effects into ASME fatigue usage factor evaluations using Equation A.20 of NUREG/CR-6909.

<u>EXPECTED COMMITTEE ACTION</u>:

The Committee is expected to provide a report recommending a course of action on this matter.

U.S. NUCLEAR REGULATORY COMMISSION March 2007

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.207

(Draft was issued as DG-1144, dated July 2006)

GUIDELINES FOR EVALUATING FATIGUE ANALYSES INCORPORATING THE LIFE REDUCTION OF METAL COMPONENTS DUE TO THE EFFECTS OF THE LIGHT-WATER REACTOR ENVIRONMENT FOR NEW REACTORS

A. INTRODUCTION

In Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," of the *Code of Federal Regulations* (10 CFR Part 50) General Design Criterion (GDC) 1, "Quality Standards and Records," requires, in part, that structures, systems, and components that are important to safety must be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function performed. In addition, GDC 30, "Quality of Reactor Coolant Pressure Boundary," requires, in part, that components included in the reactor coolant pressure boundary must be designed, fabricated, erected, and tested to the highest practical quality standards.

Augmenting these design criteria, 10 CFR 50.55a endorses the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for design of safety-related systems and components. In particular, 10 CFR 50.55a(c) requires, in part, that components of the reactor coolant pressure boundary meet the requirements for Class 1 components in Section III, "Rules for Construction of Nuclear Power Plant Components," of the ASME Boiler and Pressure Vessel Code. Specifically, those Class 1 requirements contain provisions, including fatigue design curves, for determining a component's suitability for cyclic service. These fatigue design curves are based on strain-controlled tests performed on small polished specimens, at room temperature, in air environments. Thus, these curves do not address the impact of the reactor coolant system environment on the components of the reactor coolant pressure boundary.

This regulatory guide provides guidance for determining the acceptable fatigue life of ASME pressure boundary components, with consideration of the light-water reactor (LWR) environment. In so doing, this guide describes a methodology that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable to support reviews of applications that the agency expects to receive for new nuclear reactor construction permits or operating licenses under 10 CFR Part 50, design certifications under 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," and combined licenses under 10 CFR Part 52 that do not reference a standard design. Because of significant conservatism in quantifying other plant-related variables (such as cyclic behavior, including stress and loading rates) involved in cumulative fatigue life calculations, the design of the current fleet of reactors is satisfactory, and the plants are safe to operate.

The NRC issues regulatory guides to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations, and compliance with regulatory guides is not required.

This regulatory guide contains information collections that are covered by the requirements of 10 CFR Part 50 and 10 CFR Part 52, which the Office of Management and Budget (OMB) approved under OMB control numbers 3150-0011 and 3150-0151, respectively. The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection request or requirement unless the requesting document displays a currently valid OMB control number.

B. DISCUSSION

The ASME Section III design curves, developed in the late 1960s and early 1970s, are based on tests conducted in laboratory air environments at ambient temperatures. The original code developers applied a margin of 2 on strain and a margin of 20 on cyclic life to account for variations in materials, surface finish, data scatter, and environmental effects (including temperature differences between specimen test conditions and reactor operating experience). However, the developers lacked sufficient data to explicitly evaluate and account for the degradation attributable to exposure to aqueous coolants. More recent fatigue test data from the United States, Japan, and elsewhere show that the LWR environment can have a significant impact on the fatigue life of carbon and low-alloy steels, as well as austenitic stainless steel and nickel-chromium-iron (Ni-Cr-Fe) alloys.

The staff evaluated two distinct methods for incorporating LWR environmental effects into the fatigue analysis of ASME Class 1 components. The first method involves developing new fatigue curves that are applicable to LWR environments. Given that the fatigue life of ASME Class 1 components in LWR environments is a function of several parameters, this method necessitates the development of several fatigue curves to address potential parameter variations. Alternatively, a single, bounding fatigue curve could be developed, but this approach might be overly conservative for most applications. The second method involves using an environmental correction factor (Fen) to account for LWR environments by correcting the fatigue usage calculated with the ASME "air" curves. This method affords the designer greater flexibility to calculate the appropriate impacts for specific environmental parameters. In addition, applicants have already used this method in their license renewal applications.

The NRC staff has selected the Fen method, as described in NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials."[1] In particular, Appendix A to that report describes a methodology that the staff considers acceptable to incorporate the effects of reactor coolant environments on fatigue usage factor evaluations of metal components. In addition, NUREG/CR-6909 provides a comprehensive review of, and technical basis for, the methodology proposed in this regulatory guide, including analysis of each parameter affecting the fatigue evaluations. In developing the underlying Argonne National Laboratory (ANL) models, the researchers analyzed existing data to predict fatigue lives as a function of temperature, strain rate, dissolved oxygen level in water, and sulfur content of the steel. The resultant method postulates a strain threshold, below which environmental effects on fatigue life do not occur. By definition, Fen is the ratio of fatigue life of the component material in a room temperature air environment to its fatigue life in LWR coolant at operating temperature. To incorporate environmental effects into the fatigue evaluation, the fatigue usage is calculated using ASME Section III Code provisions, and the fatigue design curve is multiplied by the correction factor.

A second concern regarding the ASME fatigue design curves involves nonconservatism of the current ASME stainless steel air design curve. More recent evaluations of stainless steel test data indicate that the ASME curve is inconsistent with the appropriate test materials and conduct of the fatigue test. Consequently, through this regulatory guide, the NRC staff endorses a new stainless steel air design curve. Section 5.1.8 of NUREG/CR-6909 provides a comprehensive review of, and technical

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[1] Copies are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone 202-512-1800); or from the National Technical Information Service (NTIS) by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161; http://www.ntis.gov; telephone 703-487-4650. Copies are available for inspection or copying for a fee from the NRC's Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555 (telephone 301-415-4737 or 1-800-397-4209; fax: 301-415-3548; email: PDR@nrc.gov). NUREG/CR-series reports are also available electronically through the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract.html.

basis for, that new design curve. The Fen defined for stainless steel in NUREG/CR-6909 should be used in conjunction with the new stainless steel air design curve when evaluating the fatigue usage of ASME Class 1 components.

In addition, the staff evaluated the incorporation of the Fen approach methodology in fatigue analyses for Ni-Cr-Fe alloys (e.g., Alloy 600 and 690) and welds. Section 6 of NUREG/CR-6909 discusses the technical basis for incorporating the environmental effects on nickel alloys and welds. In summary, fatigue evaluations for Ni-Cr-Fe alloys are based on the fatigue design curve for austenitic stainless steels. However, the existing fatigue data for Ni-Cr-Fe alloys and their welds are not consistent with the current ASME Code fatigue design curve for austenitic stainless steels. The data are either comparable or slightly conservative with the updated ANL model for austenitic stainless steels. Thus, the new fatigue design curve proposed for austenitic stainless steels adequately represents the fatigue behavior of Ni-Cr-Fe alloys and their welds. Therefore, the new design curve for austenitic stainless steels may also be used for Ni-Cr-Fe alloys and their welds. The staff finds it acceptable to use the new austenitic stainless steels air design curve in Ni-Cr-Fe alloys environmental fatigue evaluations. Consequently, Section 6 of NUREG/CR-6909 presents the respective Fen equations to be used for Ni-Cr-Fe alloys and their welds.

Section 7 of NUREG/CR-6909 evaluates the ASME design curve margins. In conducting that evaluation, researchers reviewed data available in the literature to assess the subfactors (excluding environment) necessary to account for the effects of various uncertainties and differences between actual components and laboratory test specimens. The researchers also performed statistical analyses using Monte Carlo simulations to develop fatigue design curves, using the "95/95 criterion." In other words, the curves should provide 95 percent confidence that 95 percent of the population will have a greater fatigue life than predicted by the design curves. The NRC deems this criterion acceptable because the fatigue design curves are based on crack initiation, rather than component failure, and therefore, additional margin exists between crack initiation and actual component failure. The results of the Monte Carlo simulations indicate that for both carbon and low-alloy steels and austenitic stainless steels, the current ASME Code procedure of adjusting the mean test data by a factor of 20 for life is conservative compared to the 95/95 criterion. The results also indicate that a minimum factor of 12 for cyclic life of both carbon and low-alloy steels and austenitic stainless steels will satisfy the 95/95 criterion. Figures 9, 10, and 37 of NUREG/CR-6909 present the resultant new air design curves, using margins of 12 for life and 2 for stress, for carbon steel, low-alloy steel, and austenitic stainless steel, respectively. This regulatory guide uses these new air design curves; thus, an applicant that chooses to adopt the procedure discussed in this guide to determine the fatigue life of stainless steels should use these air design curves. However, the existing ASME air design curves for carbon and low-alloy steels may also be used with the procedure in this guide to determine the fatigue life of those materials, since their use will yield conservative results.

The NRC reviewed and found acceptable several methods for calculating Fen. Only the types of stress cycles or load set pairs that exceed strain threshold criteria for carbon and low-alloy steels and austenitic stainless steels need to be considered for Fen calculations. The evaluation options depend on the complexity of the analyzed transient condition and the detail of the evaluation. For example, in an evaluation in which the results of detailed transient analyses are available to determine the necessary parameters (strain rate, temperature, and others), the "modified rate approach" (presented and referenced in Section 4.2.14 of NUREG/CR-6909) is an acceptable methodology for determining the Fen values. This methodology involves a strain-based integral for evaluating conditions for which temperature and strain rate change, resulting in variation of Fen over time. This detailed approach calculates the Fen values based on the strain history for each load set in the fatigue analysis evaluation, considering the effects of strain rate and temperature variations for each incremental segment in the strain history. Such results may be used to reduce the conservatism in the calculated Fen values. For a simplified calculation yielding a more conservative result for a complex or poorly defined set of transients, the temperature is equal to the average temperature in the transient or segment. The calculated Fen values are then used to incorporate environmental effects into ASME fatigue usage factor evaluations.

C. REGULATORY POSITION

This section describes the methods that the staff considers acceptable for use in performing fatigue evaluations, taking into account the effects of LWR environments on carbon and low-alloy steels, austenitic stainless steels, and Ni-Cr-Fe alloys. Specifically, these methods include calculating the fatigue usage in air using ASME Code analysis procedures and then employing the Fen value as described in NUREG/CR-6909. In particular, Appendix A to that report includes detailed descriptions and additional guidance concerning the overall methodology and all equations referred to in this section.

1. Carbon and Low-Alloy Steels

Use the following procedure to calculate the environmental fatigue usage of carbon and low-alloy steel components in LWR environments.

1.1 Fatigue Usage in Air

Calculate the fatigue usage in air using ASME Code analysis procedures and the fatigue air curves provided in NUREG/CR-6909, Section 4.1.10, Figures 9 and 10 (updated ANL model curves).

1.2 Environmental Correction Factor (Fen)

Calculate the environmental correction factor, Fen, using Equation A.2 of NUREG/CR-6909 for carbon steels, or Equation A.3 of NUREG/CR-6909 for low-alloy steels. Equations A.4 through A.7 of NUREG/CR-6909 should be used to calculate the respective parameters. Equation A.8 of NUREG/CR-6909 indicates the strain threshold.

1.3 Environmental Fatigue Usage

Calculate the environmental fatigue usage using Equation A.20 of NUREG/CR-6909.

2. Austenitic Stainless Steels

Use the following procedure to calculate the environmental fatigue usage of austenitic stainless steel components in LWR environments.

2.1 Fatigue Usage in Air

Calculate the fatigue usage in air using ASME Code analysis procedures and the new stainless steel fatigue air curve provided in NUREG/CR-6909, Section 5.1.8, Figure 37 (proposed design curve).

2.2 Environmental Correction Factor (Fen)

For all types of austenitic stainless steels (e.g., Types 304, 310, 316, 347, and 348), calculate Fen using Equation A.9 of NUREG/CR-6909. Equations A.10 through A.12 of NUREG/CR-6909 define the respective parameters. Equation A.13 of NUREG/CR-6909 provides the strain threshold.

2.3 Environmental Fatigue Usage

Calculate the environmental fatigue usage using Equation A.20 of NUREG/CR-6909.

3. Ni-Cr-Fe Alloys

Use the following procedure to calculate the environmental fatigue usage of Ni-Cr-Fe alloy components in LWR environments (e.g., Alloy 600 and 690).

3.1 Fatigue Usage in Air

Calculate the fatigue usage in air using ASME Code analysis procedures and the new stainless steel fatigue air curve provided in NUREG/CR-6909, Section 5.1.8, Figure 37 (proposed design curve).

3.2 Environmental Correction Factor (Fen)

For all types of Ni-Cr-Fe alloys (e.g., Alloy 600 and 690), calculate Fen using Equation A.14 of NUREG/CR-6909. Equations A.15 through A.17 of NUREG/CR-6909 define the respective parameters. Equation A.18 of NUREG/CR-6909 provides the strain threshold.

3.3 Environmental Fatigue Usage

Calculate the environmental fatigue usage using Equation A.20 of NUREG/CR-6909.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide. This regulatory guide only applies to new plants and no backfitting is intended or approved in connection with its issuance.

The methods described in this final guide reflect public comments and will be used to evaluate submittals in connection with applications for construction permits, standard plant design certifications, operating licenses, early site permits, and combined licenses.

REGULATORY ANALYSIS

The NRC staff did not prepare a separate regulatory analysis for this regulatory guide. The regulatory analysis is available in Draft Regulatory Guide DG-1144, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Environment in New Reactors," issued July 2006. The agency issued DG-1144 for public comment as the draft of this Regulatory Guide 1.207. A copy of the regulatory analysis is available for inspection and copying for a fee at the U.S. Nuclear Regulatory Commission Public Document Room, 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is US NRC PDR, Washington, DC 20555; telephone 301-415-4737 or 1-800-397-4209; fax 301-415-3548; email PDR@nrc.gov.

Tab 9

November 20, 2006

538th MEETING OF THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
PROPOSED REVISIONS TO EMERGENCY PLANNING REGULATORY GUIDE
AND REVIEW STANDARDS
DECEMBER 7, 2006
ROCKVILLE, MARYLAND

STATUS REPORT TABLE OF CONTENTS

Cognizant ACRS Members: Dr. Dana Powers, Dr. Michael Corradini

Cognizant ACRS Staff Engineer: Maitri Banerjee

538th Meeting of the
Advisory Committee on Reactor Safeguards
Proposed Revisions to EP SRP 13.3/DG-1145
December 7th, 2006
Rockville, MD

-PROPOSED SCHEDULE-

Cognizant Staff Engineer: Maitri Banerjee MXB@NRC.GOV (301) 415-6973

	Topics	Presenters	Presentation Time
I	Opening Remarks	M. Corradini, ACRS	1:15 pm - 1:25 pm
II	Proposed Revisions to SRP 13.3 - Introduction - Description of changes and technical bases - Convergence of SRP 13.3 with applicable DG-1145 sections - EP ITAAC and its use and closure - Offsite EP and impact on greenfield applications -Resolution of public comments received	NRR - Kathryn Brock - Daniel Barss	1:25 pm - 2:30 pm
III	Industry Comments	Alan Nelson, NEI	2:30 pm - 2:45 pm
IV	Full Committee Discussion	M. Corradini, ACRS	2:45 pm - 3:15 pm

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.

- 35 copies of the presentation materials to be provided to the Subcommittee.

538th MEETING OF THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
PROPOSED REVISIONS TO EP SRP 13.3/DG-1145
December 7th, 2006
ROCKVILLE, MARYLAND

- STATUS REPORT -

Emergency Planning (EP) Regulatory Guidance and Review Standard

Purpose

The purpose of this meeting session is to review proposed revisions to NUREG-0800, Standard Review Plan Section 13.3, "Emergency Planning." The Committee will hear presentations by and hold discussions with representatives of the staff and NEI. No interested public has been identified.

Previous ACRS Review

The NRC staff provided the ACRS with a draft proposed Standard Review Plan (SRP) in NUREG 0800, Section 13.3 on September 8, 2006. This complete rewrite of the SRP 13.3 incorporates the licensing processes under 10 CFR Parts 50 and 52, and has not been reviewed by the ACRS before.

Background

The draft revision to the SRP Section 13.3 was prepared by the NSIR staff in cooperation with the Department of Homeland Security (DHS) and the NRR staff to ensure up-to-date guidance is available for the staff to review applications for new sites and reactors. The regulatory requirements for an EP program, as codified in Part 50 remain basically unchanged. The most significant change to the SRP is incorporation of the Part 52 application process. The proposed revision was issued for public comments on September 30, 2006, and the comment period expired on November 13, 2006. Several comments were received via NEI from the industry.

In SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)," October 28, 2005, the staff proposed the use of generic EP ITAAC as a model for inclusion in Combined License (COL) applications. The generic EP ITTAC represents a first-of-a-kind example of programmatic ITAACs under Part 52 that was developed working with NEI and the DHS/FEMA. The staff concluded that an applicant needs to submit EP ITAAC as part of a complete and integrated emergency preparedness program description with the COL application for the staff to be able to arrive at a reasonable assurance finding. The ITAAC should be limited to those aspects of emergency planning and preparedness that cannot be addressed reasonably prior to construction of the plant. This is reflected in the proposed new Part 52 rulemaking which also allows an ESP applicant to use the EP ITAAC in association with the option to provide complete and integrated emergency plans at the ESP stage. Allowing the inclusion of emergency preparedness ITAAC in the ESP application is consistent with the

Commission's goal of resolving siting issues early in the licensing process. The ITAAC would need to be completed before loading fuel in the reactor. In a Staff Requirements Memorandum dated February 22, 2006, the Commission approved the generic EP ITAAC as a minimum set for EP program to be included in a COL application, recognizing that the acceptability of the proposed plant specific EP ITAAC would be reviewed on a case-by-case basis.

Summary of Proposed Revisions to SRP 13.3

The proposed revision to the SRP identifies the acceptance criteria for applications submitted under 10 CFR Parts 50 and 52 by adding references to the applicable sections of the 10 CFR. The regulatory guidance provided in the SRP references various applicable NUREG documents (including NUREG-0654/FEMA-REP-1, Rev. 1, NUREG-0696, NUREG-0737) and industry guidance (NEI 99-01) with exceptions clarified. In addition to including the Commission policy on the use of ITAAC at the COL stage, the staff provides an option of using the EP ITAAC at the ESP stage consistent with the ongoing Part 52 rulemaking. Experience gained from the review of the first three ESP applications and the standard design certification processes are incorporated.

The SRP addresses non-participation by the State and local governments. A full-participation exercise need not be performed before issuance of a COL. However, in accordance with proposed Part 52 rulemaking, an exercise is required before fuel loading. Operation up to 5% power level would be allowed while offsite deficiencies identified during the exercise were resolved. As the ITAACs need to be completed before fuel loading, the staff has also proposed a license condition to be added to all COLs allowing operation at up to 5 percent power with deficiencies identified by FEMA. In SECY-05-0203, "Revised Proposed Rule to Update 10 CFR Part 52, Licenses, Certifications, and Approvals for Nuclear Power Plants," November 3, 2005, the staff expressed concern that allowing COL holders to operate at up to 5 percent power with offsite emergency preparedness deficiencies could result in consequent contamination of the reactor with no assurance that the plant will ever operate at full power (e.g., Shorham-like scenario).

Consistent with the proposed Part 52 rulemaking, if an application identifies a significant impediment to the development of emergency plans due to physical characteristics unique to the proposed site or an evacuation time estimate study that show such an impediment, the staff guidance in the SRP states that the application must identify measures that would, when implemented, mitigate or eliminate the significant impediment.

The staff also proposes to add new requirements to Part 50 if the applicant has an operating reactor at the site. An exercise, either full- or partial-participation, needs to be conducted for each subsequent reactor constructed on the site. With regard to subsequent reactors, those aspects of an exercise which address currently untested (i.e., unexercised) aspects of emergency preparedness for the proposed new reactor must be addressed in the new emergency preparedness ITAAC.

For a COL applicant referencing an ESP, the staff is proposing that an updating requirement be imposed. This includes new information and inaccuracies in the emergency preparedness information that may materially affect the Commission's earlier determination on emergency preparedness reasonable assurance finding. In addition, the staff is proposing that the applicant must discuss whether the new information could materially change the bases for compliance with the applicable NRC requirements.

The proposed revision to the SRP also identifies a new report NUREG/CR-6863, "Development of Evacuation Time Estimate Studies for Nuclear Power Plants," that updates the existing guidance by integrating new technology tools in traffic management, computer modeling, and communication systems to estimate the evacuation times.

The staff is continuing to work on finalizing three major areas, which are as follows:

> -The minimum amount of offsite information an applicant needs to provide for offsite EP to allow the staff to make a reasonable assurance finding (primarily a greenfield site issue). The staff continues to work with DHS/FEMA;
>
> -Finalizing the list of generic EP ITAACs from experience gained through the review of first such ITAACs submitted with the Vogtle ESP application;
>
> -Experience from the review of Vogtle ESP application (the first applicant to submit a complete and integrated emergency preparedness plan) will be factored into the ESP review, as needed.

Additionally, the ongoing Part 52 rulemaking, when finalized, could impact the SRP.

Feedback from Industry

Nuclear Energy Institute (NEI) submitted comments on the proposed revision of the SRP in a letter dated November 9, 2005. Following the theme of their comments on the Part 52 rulemaking, NEI's comments identify the following concerns:

1. Application for a new reactor at an existing site should not open the existing site emergency plan for review. The new plan should stand on it's own unless the planning standard is interlinked and common to both the existing reactor and the new reactor.

2. The SRP should not expand on the base set of Generic Emergency Planning Inspections, Tests, Analysis, and Acceptance Criteria as provided in SECY-05-0197.

3. The use of the term "generic communications" is inconsistent with requirements in proposed Part 52.79(a)(37) which limits this scope to bulletins and generic letters.

4. There is no regulatory basis or precedent requiring the submittal of offsite implementing procedures. Offsite implementing procedures historically are evaluated as part of the biennial exercise.

5. There is a concern regarding the absence of DHS/FEMA/REP, planning references and limited offsite emergency response plan related review criteria.

The staff is currently reviewing and resolving these comments.

Preliminary Questions from the ACRS on Draft SRP 13.3 Provided to the Staff

The following draft questions were provided to the staff to help them prepare for the Full

Committee briefing:

1. Although this is a complete rewrite of the SRP the only substantive addition that we could see was the addition of the 10 CFR 52 process. Is that accurate? If not, can you identify other substantive changes?

2. Additional revisions may come for application of SRP 13.3 to address 'greenfield' sites since there is no substantive guidance to the offsite planning information needed in support of a staff finding for this. Please discuss the staff's plan for this situation.

3. The EP- ITAAC Table may not be complete for use as an ESP. We note the use of ITAAC Table is optional at the ESP stage, and may change after public comments. Please elaborate the staff's thinking in this area.

4. NUREG-0654 guidance for evacuation plans seems to be limited in addressing the regulation that requires a diversity of emergency plans (full range of options including evacuation, sheltering and possible KI usage). Does the new SRP 13.3 address a proper review of the needed diversity of planning options? If yes, please elaborate how it is done.

5. Please discuss briefly disposition of significant comments received from the public

Expected Committee's Action

The Full Committee will gather information, analyze relevant issues and facts, and formulate proposed positions and recommendations, as appropriate, and prepare a letter on the proposed SRP and applicable portions of DG-1145.

References

1) Memorandum from David B. Matthews to John Larkins, Transmittal of Proposed Draft Revision to Standard Review Plan, NUREG-0800, Section 13.3, "Emergency Planning," September 8, 2006 (ML061870206).

2) Commission Paper SECY-05-0197, Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria, October 28, 2005 (ML052770225), and the related Staff Requirement Memorandum dated February 22, 2006 (ML060530316)

3) Commission Paper SECY-05-0203, "Revised Proposed Rule to Update 10 CFR Part 52, Licenses, Certifications, and Approvals for Nuclear Power Plants," November 3, 2005 (ML052300372), and the related Staff Requirement Memorandum dated January 30, 2006 (ML060300640).

4) NEI Letter dated November 9, 2006, NUREG-0800 Section 13.3, "Emergency Planning" Request for Comment.

NUCLEAR ENERGY INSTITUTE

Alan P. Nelson
DIRECTOR EMERGENCY
PREPAREDNESS
NUCLEAR GENERATION

November 9, 2006

Chief, Rulemaking, Directive, and Editing Branch
U.S. Nuclear Regulatory Commission
Mail Stop T6-D59
Washington, DC 20555-0001

Project 689

SUBJECT: NUREG-0800 Section 13.3 "Emergency Planning"
Request for Comment

On behalf of the nuclear industry, the Nuclear Energy Institute (NEI)[1] is pleased to submit the following response to The Federal Register, dated September 29, 2006, Volume 71, Number 189 which invited written comments on Section 13.3, Second Draft Revision 3, "Emergency Planning" of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition".

The New Plant Emergency Preparedness Task Force has identified the following significant concerns:

- Application for a new reactor at an existing site should not open the existing site emergency plan for review. The new plan should stand on it's own unless the planning standard is interlinked and common to both existing reactor and new reactor.
- The SRP should not expand on the base set of Generic Emergency Planning Inspections, Tests, Analysis, and Acceptance Criteria as provided in SECY-05-0197.
- The use of the term "generic communications" is inconsistent with requirements in proposed Part 52.79(a)(37) which limits this scope to bulletins and generic letters.
- There is no regulatory basis or precedent requiring the submittal of offsite implementing procedures. Offsite implementing procedures historically are evaluated as part of the biennial exercise.
- There is a concern regarding the absence of DHS/FEMA/REP, planning references and limited offsite emergency response plan related review criteria.

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[1] NEI is the organization responsible for establishing unified industry policy on matters affecting the nuclear energy industry. NEI's members include all entities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect/engineering firms, fuel fabrication facilities, nuclear material licensees, and other organizations and individuals involved in the nuclear energy industry.

Attachment 1 provides specific comments and recommendations. Attachment 2 discusses the use and application of Generic Emergency Planning Inspections, Tests, Analysis, and Acceptance Criteria as provided in SECY-05-0197.

We appreciate the opportunity to comment on the draft document. Once you have had an opportunity to review the attached recommendations, we would like to schedule a meeting. If you have any questions regarding this effort please contact Marty Hug by e-mail mth@nei.org or phone 202-739-8129.

Sincerely

Alan Nelson

cc: NRC Document Control Desk
Nader Mamish

Enclosures

Attachment 1

Comments on NUREG-0800 Section 13.3

	Section	Page	Line	Numbered Paragraph	Comment and Recommended Actions
1	I	13.3-1	17		10 CFR 73.1 should be addressed in the Security Plan not the Emergency Plan. Remove requirement.
2	I	13.3-2	2		The guidance provided in the NUREG-0800 and NUREG-6863 regarding the Evacuation Time Estimate (ETE) and the consideration that is given to the construction force does not provide sufficient detail. The ETE guidance should provide clear direction to the reviewer (and subsequently) to the applicant regarding the scope. For example, the statement provided in the NUREG-0800 does not clearly establish the need to address the construction workforce that could be years down the road, but the inclusion of that number of personnel could skew the results of the ETE significantly and therefore, alter the proper Protective Action Recommendations for the personnel in the EPZ. There is a lack of guidance in the document regarding what is expected to support maintaining the size and shape of the Emergency Planning Zone and what factors must be taken into account to ensure that the reviewer has all the pertinent information to make a determination that the EPZ is adequate to support the building and operation of an additional Unit(s) on a particular site. Provide additional guidance.
3	I	13.3-2	27		Line 27 states, "*The review addresses such areas as a habitable technical support center (TSC) with adequate space, data retrieval capabilities and dedicated communications equipment, and an operational support center (OSC) with adequate communications.*" This statement assumes that the TSC will be "within a two minute walk to the Control Room." Technology advancements in onsite communications do not support this review criterion. The industry plant development teams are recommending a single stand alone TSC. The guidance should be revised to allow for a TSC that is not within 2 minutes of the control room.
4	I	13.3-2	36		Change 10 CFR 52.80 to 10 CFR 52.81
5	I	13.3-3	27		Paragraphs 4 and 5 state that NRC consults DHS's review of offsite plans and preparedness. However no guidance is provided in the SRP as to the extent of the review by FEMA. Provide guidance.
6	I	13.3-3	29	4	Editorial Comment – Change DHS back to FEMA. DHS is used in a number of places in document.

Attachment 1

Comments on NUREG-0800 Section 13.3

	Section	Page	Line	Numbered Paragraph	Comment and Recommended Actions
7	II	13.3-5	3		Editorial Comment: Under section II. Acceptance Criteria, the "lettering" begins with L, M, N... This should be A, B, C...
8	II	13.3-6	7		Editorial Comment: Under Regulatory Guidance, numbering should start with 1 and 2, rather than 12 and 13.
9	II	13.3-7	1	3	Sentence states that the applicant should use NEI 99-01 Revision 4. The SRP should reference that Revision 4's Security EALs were modified by Bulletin 2005-02 and the NEI white paper endorsed in RIS 2006-12. Also recognize that Revision 5 is in process and will be updated and include Security EALs.
10	II	13.3-7	16	3	"Emergency actions" should be changed to "emergency action levels."
11	II	13.3-7	18	3	Change "emergency plan" to "submittal."
12	II	13.3-8	9	8	Referring to multiple revisions of Reg. Guide 1.101 is confusing given the purpose of Reg. Guides to provide an acceptable method for compliance. The industry recommends revising RG 1.101 to accommodate all the acceptable methods rather than relying on 3 or 4 different versions.
13	II	13.3-8	34	11	The first sentence of paragraph 11 states "...application for an OL or COL provide an analysis..." Cited regulations do not require that Evacuation Time Estimate be submitted to the NRC as part of the COL application. Change provide to perform to make it clear that the ETE is not submitted with the COL.
14	II	13.3-10	40	17	Editorial Comment: NUREG 654 should be NUREG 0654
15	II	13.3-14	18	29	NUREG 1022. Add "revision 2 as per reference #47."
16	III	13.3-17	19	2	Change "...separate document identified as..." to "...separate document referenced by....."
17	III	13.3-18	8	4	Last sentence of the paragraph refers to a review of recent NRC emergency planning and health physics reports. This should be removed since it has no basis as part of a review for a new license application.

Attachment 1

Comments on NUREG-0800 Section 13.3

	Section	Page	Line	Numbered Paragraph	Comment and Recommended Actions
18	III	13.3-18	29	6	Editorial Comment: "Residences" should be "Residents
19	III	13.3-19	1	8	Make following revision to the first and second sentence – "*In general, if an applicant for an additional reactor at an operating reactor site, and the applicant proposes to incorporate and extend elements of the existing emergency planning program to the new reactor (included by reference), ~~those existing elements should be considered acceptable and adequate.~~ the reviewer should generally focus the review on the extension.....*". Application for a new reactor at an existing site should not open the existing site emergency plan for a review of commitments. The elements that are extended to the new plan should stand on their own, when possible unless the element is interlinked in such a way as the element is common to both existing reactor and new reactor.
20	III	13.3-20	5	10	Insert into the third sentence "*The reviewer should identify any deficiencies, cite the regulatory basis, and use.....*"
21	III	13.3-20	19	11	RAIs should include a reference citing the applicable requirements they are related to. This will help the applicant better understand the NRC staff's question/concern and allow the applicant to more effectively respond to the RAI.
22	III	13.3-21	17	14	The use of the term "generic communications" is inconsistent with requirements in proposed Part 52.79(a)(37) which limits this scope to bulletins and generic letters. Revise this section to be consistent with 52.79(a)(37)
23	III	13.3-21	31	16	Reporting requirements for safeguards events are covered by the standard emergency classification and action level scheme discussed in paragraph 3 on page 13.3-6. This paragraph does not introduce a new requirement. Paragraph 16 should be removed or reference page 13.3-6 paragraph 3..
24	III	13.3-21	31	17	State the regulatory basis for this requirement.
25	III	13.3-25	33	2	The regulatory requirements for submitting implementing procedures are addressed in Part 50 and do not require submitting implementing procedures with the COL application. Site implementing procedures are also addressed under ITAAC 15.1. State/local procedures are tested during the evaluated exercise. ITAAC 12.1.3

Attachment 1

Comments on NUREG-0800 Section 13.3

	Section	Page	Line	Numbered Paragraph	Comment and Recommended Actions
					addresses this requirement.
26	III	13.3-27	8	8	The additional ITAAC in Table 13.3-1 ITAAC that are not "* & **bolded** text" are inappropriate and should be deleted. See ITAAC discussion in Attachment 2.
27	IV	13.3-32	11		Remove reference to 10 CFR 73.71. This should be a Security Plan reference.
28	IV	13.3-33	4	b	Paragraph should reference NUREG 0696 instead of RG 1.101.
29	IV	13.3-33	10	c	RG 1.101 does not reference NUREG-0696. Correct reference.
30	IV	13.3-33	18	d	RG 1.101 does not discuss habitability. Correct reference.
31	IV	13.3-34	10		10 CFR 73.1 should be addressed in the Security Plan not the Emergency Plan.
32	IV	13.3-35	6		10 CFR 73.1 should be addressed in the Security Plan not the Emergency Plan.
33	VI	13.3-35	28		No references are provided FEMA documents that will be used to review the submittal. Provide references.
34	Table 13.3-1	13.3-46			Add column headings to each table page.
35	Table 13.3-1	13.3-46	13	Acceptance Criteria 4.1	Revise sentence: The test would be performed using a simulated emergency.
36	Table 13.3-1	13.3-48	4	Acceptance Criteria 7.1.2	Editorial Comment: "*Advanced communications capabilities may be used ~~to satisfy~~ in lieu of the two minute travel time.*"
37	Table 13.3-1	13.3-49	1	Acceptance Criteria 7.1.6	Revise sentence: "*The OSC is located onsite, separate from the control room ~~and TSC~~.*
38	Table 13.3-1	13.3-53	7	EP program element 9.1	Numbering of elements below 9.1 is not correct.
39	Table 13.3-1	13.3-54	1	E,T and A 10.1 to 10.4	Change test to inspection.
40	Table 13.3-1	13.3-56	1	Acceptance Criteria 13.1	Change test to inspection.

Attachment 2

Comments on NUREG-0800 Section 13.3 - ITAAC

COL applicants are required to submit complete and integrated emergency plans with their applications. SECY-05-0197 documents the set of EP ITAAC established based on extensive stakeholder interactions that are to be submitted along with complete and integrated emergency plans consistent with Part 52 requirements. Part 52 also provides the option to submit complete and integrated emergency plans as part of an ESP application, and proposed Section 52.17(b)(3) requires EP ITAAC to also be provided under that option. The new requirement for EP ITAAC is based on the logic that the NRC staff needs the same information to approve complete and integrated emergency plans whether the plans are submitted at the ESP or COL stage.

Given this logic, the purpose of the additional (un-bold, un-starred) ITAAC in Table 13.3-1 is not clear. We expect that EP ITAAC for complete and integrated emergency plans to be the same (the ones identified in SECY-05-0197) whether the plans are submitted with an ESP application or COL application. In a public meeting on Oct. 21, the staff explained that the additional EP ITAAC were intended for use by an ESP applicant whose EP information is incomplete in one or more respects. This approach may provide valuable flexibility in some future, as yet unforeseen circumstance, and we do not object to retaining this option in the SRP. However, we recommend that the proposed additional (un-bold, un-starred) ITAAC not be identified in Table 13.3-1. It is not necessary to do so because such ITAAC can be developed on a case basis in the future. Moreover, removing them from the SRP will avoid confusion on the part of future industry and NRC staff regarding their regulatory status and purpose.

Apart from these general concerns, there are problems with the specific additional ITAAC proposed in the SRP for potential use by ESP applicants. Examples include:

- Proposed additional ITAAC 1.1 states "An inspection of implementing procedures or staffing rosters will be performed." This ITAAC does not address a lack of information at ESP that will not also exist at the COL stage. Staffing rosters would not be available to submit with either ESP or COL applications. The need for an ITAAC in this area was considered in the development of SECY-05-0197 and was not determined to be necessary.
- Based on 10 CFR 50.47(b)(11), proposed additional ITAAC 10.0, Radiological Exposure Control, requires a test be performed of the capabilities for onsite radiation protection. The need for an ITAAC in this area was considered in the development of SECY-05-0197 and was not determined to be necessary. Onsite radiation protection capabilities will be demonstrated as part of the on-site exercise (EP ITAAC 12.1).
- Proposed additional ITAAC 8.6 reads, "The means exists for field monitoring within the plume exposure EPZ." This capability is inherent in required (bold, starred) ITAAC 8.1 which states, "The means exist to provide initial and continuing radiological assessment through the course of the accident."

Attachment 2

Comments on NUREG-0800 Section 13.3 - ITAAC

Consistent with the approach used in developing SECY-05-0197, ITAAC 8.6 is not needed.

- Under Element 9.0, Protective Actions proposed additional ITAAC 9.2 reads, "The means exist to radiologically monitor people evacuated from the site." This ITAAC has nothing to do with protective actions developed for the plume exposure emergency workers. An ITAAC in this area was not determined to be necessary as part of the development of SECY-05-0197.

These problems reflect that the additional (un-bold, un-starred) ITAAC proposed have not had the benefit of substantial stakeholder discussions similar to the year-long interactions that led to the required EP ITAAC identified in SECY-05-0197. They are not needed, are not consistent with the principles that guided development of SECY-05-0197, and should be removed from the SRP. For consistency, the additional proposed (un-bold, un-starred) EP ITAAC should also be deleted from Section C.II of DG-1145.

Tab 5

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
538th MEETING
STATE-OF-THE-ART CONSEQUENCE ANALYSES PROJECT
December 7, 2006

-TABLE OF CONTENTS-

Cognizant ACRS Member: William J. Shack
Cognizant ACRS Staff Engineer: Hossein P. Nourbakhsh

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
538th MEETING
STATE-OF-THE-ART CONSEQUENCE ANALYSES PROJECT
December 7, 2006

-PROPOSED SCHEDULE -

SUBJECT	PRESENTER	TIME
I. Introductory Remarks and Objectives	William Shack	3:30 - 3:35 P.M.
II. NRC Staff Presentation	RES staff	3:35 - 5:10 P.M.
III. Committee Discussions		5:10 - 5:30 P.M.

Note: Presentation time should not exceed 50% of the total time allocated for a specific item.
Number of copies of presentation materials to be provided to the ACRS-40

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
538th MEETING
STATE-OF-THE-ART CONSEQUENCE ANALYSES PROJECT
December 7, 2006

-STATUS REPORT-

PURPOSE

The purpose of this meeting is to discuss the status of the staff's efforts associated with the state-of-the-art reactor consequence analysis project.

BACKGROUND AND DISCUSSIONS

The phenomenology and offsite consequences of severe reactor accidents has been the subject of considerable research by the NRC. Over the years, several systematic attempts has been made to use quantitative techniques to estimate the probabilities, source terms, and public consequences from potential accidents in commercial nuclear power plants. The Reactor Safety Study (WASH-1400), was the first systematic attempt to provide estimates of public risk. This 1975 study included analytical methods for determining both the probabilities and consequences of various accident scenarios. Two specific reactor designs were analyzed in WASH-1400: Peach Bottom Atomic Power Station, a Boiling Water Reactor (BWR) with a Mark I containment and Surry, a 3-loop Pressurized Water Reactor (PWR) with a subatmospheric containment.

Sandia National Laboratory (SNL) performed a study of technical aspects of siting for nuclear power reactors. The results of this study, also known as Sandia Siting Study, were published in NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," December 1982. This study used five generic source terms for analyzing the consequences and socio-economic impacts of possible plant accidents at 91 existing or proposed reactor sites. These source terms were derived from the Reactor Safety Study (WASH-1400) and its immediate successors.

Since the publication of the Sandia Siting Study, many events have brought a new focus to this study and its results. The results, in terms of predicted offsite early fatalities and latent cancer , have often been quoted by outside organizations to illustrate the potential consequences of a severe accident at a commercial nuclear power plant. Despite accepted arguments that these results does not present an up-to-date picture of consequences at nuclear power plants and does not reflect current state-of-the-art in evaluating severe accident progression and offsite consequences.

On request from the Commission, the staff sent forward to the Commission a paper describing a proposed plan for developing state-of-the-art reactor consequence analyses for all commercial nuclear power plant sites. The Commission responded in an April 14, 2006 Staff Requirements Memorandum (SRM) with a general approval of the plan. The Commission directed the staff to "use the improved understanding of source terms and severe accident phenomenology (e.g., containment failure modes, time of release, release duration, inventory release fractions), and credit the use of Severe Accident Management Guidelines (SAMGs) and other new procedures, such as mitigative measures resulting from B.5.b and other like programs, that were not in place when the earlier study was performed." The Commission also instructed the staff to "present its updated results using risk communication techniques to achieve an informed public understanding of the extent and value of defense-in-depth features including current mitigative strategies, and of the important analytical assumptions."

In the SRM, the Commission specifically instructed the staff to "work with the ACRS on technical issues such as identification of accident scenarios to be evaluated, evaluation of source terms, credit for operator actions or plant mitigation systems, modeling of emergency preparedness, modeling of offsite consequences, and definition and characterization of analysis uncertainty."

During the 535th meeting of the ACRS, September 7-9, 2006, the staff briefed the Committee on its plan for the state-of-the-art consequence analyses project . The purpose of this meeting is to discuss the current status of the staff's efforts associated with this project including MACCS code improvement and the communication plan.

EXPECTED COMMITTEE ACTION

The Committee is not expected to issue a letter at this time.

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December 22, 2005 SECY-05-0233

FOR: The Commissioners

FROM: Luis A. Reyes
Executive Director for Operations

SUBJECT: PLAN FOR DEVELOPING STATE-OF-THE ART REACTOR CONSEQUENCE ANALYSES

PURPOSE:

The purpose of this paper is to inform the Commission of the staff's plan incorporating the combined efforts of the Offices of Nuclear Regulatory Research (RES), Nuclear Reactor Regulation (NRR), and Nuclear Security and Incident Response (NSIR) to (1) evaluate and update, as appropriate, analytical methods and models for realistic evaluation of severe accident progression and offsite consequences; (2) develop state-of-the-art reactor consequence assessments of severe accidents and update such analyses as NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," dated December 1982; (3) identify mitigative measures that have the potential to significantly reduce risk or offsite consequences; and (4) develop an integrated, faster than real-time, computer-based tool to assist decision-making in the event of a severe reactor accident. This paper does not create any new commitments.

DISCUSSION:

The evaluation of accident phenomena and offsite consequences of severe reactor accidents has been the subject of considerable research by the U.S. Nuclear Regulatory Commission (NRC). Most recently, with Commission guidance and as part of plant security assessments, the staff has concentrated on applying the accumulated research to perform analyses of severe accident progression and consequences, which are considerably more detailed, integrated, and realistic than past analyses. The results of these recent studies have confirmed and quantified what was suspected but not well-quantified — namely, that some past studies of plant response and offsite consequences could be extremely conservative, to the point that predictions were not

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useful for characterizing results or guiding public policy. In some cases, the overly conservative results were driven by the combination of conservative assumptions or boundary conditions; in other cases, simple bounding analysis was used in the belief that if the result was adequate to meet an overall risk goal, bounding estimates of consequences could be tolerated.

The subsequent misuse or misinterpretation of such bounding estimates further suggests that communication of risk attributable to severe reactor accidents should be based on realistic estimates of the more likely outcomes.

The staff is planning to perform consequence analysis for scenarios of radiological release frequency greater than or equal to 10^{-6} per reactor year. If there are important security related events which are not captured by the spectrum of scenarios adopted for safety analysis, the staff, with Commission approval, can analyze those scenarios as part of the classified version of the study report. The analyses of such security related events should focus on additional mitigation as well as unmitigated consequences.

The staff has developed the attached plan to create a body of knowledge regarding the likely outcomes of severe reactor accidents, based on the most current emergency preparedness (EP) and plant capabilities and to identify reasonable and efficacious means by which to further mitigate such events. Through the evaluation of best available modeling and uncertainties, the staff also anticipates identifying opportunities for further efficient improvement and validation of modeling.

The basic approach will be to utilize the integrated modeling of accident progression (reactor and containment thermal-hydraulic and fission product response), which is embodied in the MELCOR code, coupled with modeling of offsite consequences (MACCS code) in a consistent manner (e.g., accident timing), drawn from our recent security assessments, to estimate offsite consequences for important classes of events. Toward that end, the staff will select events with appropriate consideration of probability. The staff will also perform offsite consequence analyses on a site-specific basis (reflecting site-specific population distributions and EP), although general accident progression modeling will be based on plant groupings by reactor and containment design types. In implementing this approach, it will be important to reflect all of the system and procedural plant improvements that have been incorporated as part of the industry's response to the NRC's security initiatives. Some additional analyses may also be needed to capture plant design specificities that would bear on severe accident probabilities or plant response.

The staff expects that the results of the reanalysis of severe accident consequences would provide the foundation for communicating that aspect of nuclear safety to Federal, State and Local authorities, licensees; and the general public. This evaluation of severe accident consequences would also update and replace the site-specific quantification of offsite consequences found in NUREG/CR-2239 and NUREG/CR-2723, "Estimates of the Financial Consequences of Reactor Accidents," dated September 1982. Publicly issued documents must incorporate effective risk communication and will be peer reviewed to ensure that objective is met.

The development and application of an integrated, realistic methodology for use in assessing the consequences of hypothetical severe accident scenarios at individual reactor sites would also benefit the NRC's response to any real future events. The NRC's Operations Center does not currently have the capability to evaluate developing reactor scenarios using faster than real-time accident progression analysis directly coupled with consequence analysis.

Consequently, the Operations Center currently evaluates offsite projected doses using a generic radiological release (based on NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," dated February 1995), which is then adjusted (in a simplified way) based on available knowledge of containment status and systems operation. The staff concludes that the same modeling techniques used in developing state-of-the-art reactor consequences (as described in the enclosure to this paper) can also be used to enhance the NRC's capability to respond to real events and assist in training NRC personnel in preparing for such events. As part of the enclosed plan, the staff proposes an activity to develop a faster than real-time version of the coupled MELCOR and MACCS codes, which the Operations Center could use in evaluating reactor events. This would afford the capability to project the timing and progression of key events (e.g., steam generator level and boiloff, core water level and uncovery, fission product release) and the alteration of the progression as a result of systems recovery and intervention. Offsite consequence estimates (dose projections, health effects, land contamination and costs) would also be available to decision-makers to further guide emergency response.

The overall schedule for this work will span approximately 3 years; however, selected higher-priority work will be scheduled for completion within the first year. Estimates (and documentation) of consequences for selected high-population and other reactor sites will be targeted for December 2006. Analyses to support development of preliminary design criteria for additional mitigation of offsite releases (i.e., beyond readily available measures) will be completed by May 2006. Analyses to evaluate the benefits of those additional active mitigation measures for specific accident scenarios will be completed by October 2006.

RESOURCES:

The staff estimates that the resource requirements of the proposed plan will be $7.45M and 12 FTE spread over 3 years, if the project is fully funded starting in FY 2006.

For RES, the resource requirements are $3,050K in FY 2006, $2,700K in FY 2007 and $1,700K in FY 2008, as well as 3.0 FTE per year.

For NRR, the resource requirements are 0.5 FTE in FY 2006, 0.25 FTE in FY 2007 and 0.25 FTE in FY 2008. None of the NRR resources are currently budgeted.

NSIR intends to support the project in FY2006 within existing budgeted resources. The combined 2 FTE required in FY 2007 and FY 2008 are not currently budgeted.

In FY 2006 RES has $229K and 2.4 FTE in the budget, and $2,821K and 0.6 FTE unbudgeted.

Unbudgeted resource requirements for RES and NRR in FY 2006 will be addressed through reallocation of lower priority work outside of new reactor licensing or, if necessary during the mid-year resource review process.

If funding is not available from mid-year, the staff is considering the list below of lower priority RES activities that may be displaced, deferred, or canceled, in order to fund the proposed plan.

- safety margin related to steam generator tube integrity,
- specific activities of the Generic Safety Issues Program,
- development and assessment of thermal-hydraulic tools,
- development and assessment of reactor fuel tools,
- development and assessment of containment and severe accident tools,
- support for licensing of mixed oxide fuel facility,
- reactor oversight process support,
- human factors research and regulatory support,
- materials aging models for passive component risk, and
- advancements in structure and earthquake engineering.

Detailed impacts of the portions that will be displaced, deferred, or canceled will be provided by a memorandum to the Commissioners in accordance with the agency's implementing procedures on reporting resource reallocations to the Commissioners.

In FY 2007, RES has budgeted $1,500K and 3.0 FTE. Additional FY 2007 and FY 2008 resource needs for RES, NRR, and NSIR will be addressed in the FY 2008 Planning, Budget, and Performance Management process.

At the December 12, 2005, Closed Commission Meeting on security-related research, the Commission inquired as to how the staff could use additional funding to facilitate the development of this project. If contract funding becomes available, such funding would most effectively be used to minimize or eliminate the impact on the projects identified above for displacement, deferral or cancellation, and to initiate testing of the most promising strategies for additional offsite radiological release mitigation (e.g., area sprays for aerosol scrubbing). The current proposed program has experimental validation of such beyond readily available measures as a future, unfunded activity.

RES will have overall responsibility for project management and coordination, technical direction and support, and review. NRR will support scenario selection and probabilistic quantification, and will provide site-specific information, as needed. NSIR will provide technical direction and support for EP modeling, and will provide plant-specific information regarding security enhancements related to plant system and procedural modifications. Senior management oversight will be provided through an agency-level steering committee.

COORDINATION:

The Office of the General Counsel reviewed this package and has no legal objection. The Chief Financial Officer reviewed this package and determined that it has no financial impact.

/RA/

Luis A. Reyes
Executive Director
for Operations

Enclosure:
Plan to Develop State-of-the-Art Consequence Analyses

PLAN TO DEVELOPING STATE-OF-THE-ART CONSEQUENCE ANALYSES

Goals

Assess the realistic consequences of a spectrum of risk-significant radiological releases to support safety- and security-related decision-making and to update such analyses as NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," dated November 1982, which Sandia National Laboratory developed for the U.S. Nuclear Regulatory Commission (NRC).

Objectives

Using a methodology based on state-of-the-art analytical tools, determine "best estimates" of the radiological dose consequences (including early and latent fatalities and land contamination) for each U.S. operating reactor site and present those results using risk communication techniques to achieve informed public understanding of the following factors:

- the extent and value of defense-in-depth features of plant design and operation, including mitigative strategies that are employed to reduce risk
- the most significant influential assumptions

As a starting point, the methodology to be used will reflect currently existing analytical research tools, coalesced into an integrated and coherent methodology to predict realistic outcomes. These analytical tools will be reviewed to identify potential substantive and cost-effective improvements that can be implemented in a timely manner, and those improvements will be incorporated. The staff will also identify potentially cost-beneficial areas for experimental validation using a systematic (internal NRC) process to identify the key influential analysis variables and assumptions, evaluate the degree of uncertainty associated with each, and determine the degree of cost and difficulty associated with reducing those uncertainties. Toward that end, the staff will use a catalog of available quantitative uncertainty results and other assumptions as the basis for identifying candidates for validation. The staff will also develop a research plan to guide continued substantive improvement, where possible, to the technical defensibility of the analytical tools developed and used in this study.

In addition, the staff will develop an integrated faster than real-time, computer-based decision-making tool, which can be used to enhance NRC and Federal responses to events of national significance. Toward that end, the staff will obtain input from computer code modeling efforts conducted by the Illinois Emergency Management Agency (IEMA).

Enclosure

Potential Regulatory Uses Include:

1. Improved Regulatory Analyses
   a. backfitting decisions
   b. rulemaking
   c. prioritization and resolution of generic safety issues
   d. identification of safety issues
   e. resource allocation through the Planning, Budgeting, and Performance Management (PBPM) process

2. New and advanced reactor licensing and siting reviews

3. Emergency preparedness (EP) to assess the effectiveness of emergency action levels (EALs) and resolution of timing issues

4. Assessment of the effectiveness of proposed security mitigation strategies

5. Better informed public dialogue (with Federal, State and local authorities; licensees, and the public) on
   a. reactor safety issue resolution
   b. security issues assessment
   c. new reactor design and siting reviews

6. Improved insights into licensees' current EP evacuation and sheltering strategies

7. To inform NRC's recommendations to DHS for beyond-readily-available mitigation strategies

Summary

The NRC staff has developed this plan to generate realistic release and consequence analyses for all nuclear power plant sites in the United States. To support more realistic and risk-informed regulatory decision-making, we will truncate scenarios that are considered extremely unlikely, so as not to obscure the value of preventive and mitigative features for the more likely scenarios. Hence, we will conduct consequence analyses only for scenarios that have a radiological release frequency (due to containment failure or containment bypass) greater than or equal to 10^{-6} per reactor year. In addition, consistent with the Commission's direction in SRM dated May 14, 2003, the latent health effects analyses will cover a range of dose models. Specifically, the selected range will encompass models with thresholds from 0 to 5 Rem.

Studies such as the NRC's Individual Plant Examination (IPE), Individual Plant Examination of External Events (IPEEE), and Simplified Plant Analysis Risk (SPAR) programs, as well as the Risk Analysis and Management for Critical Asset Protection (RAMCAP) study[1] conducted by the Electric Power Research Institute (EPRI), yield generic risk insights. Consequently, in developing state-of-the-art release and consequence analyses for all nuclear power plant sites in the United States, we will use the generic risk insights from these and other studies to identify generic accident scenarios. We will then calculate generic source terms and consequences for those analyses (using existing analyses where appropriate). Alternatively, we may use fuel damage classes (such as those identified in RAMCAP) to risk-inform the full spectrum of potential fuel damage classes that we will consider. We will also need to screen those fuel damage classes to focus on risk-significant scenarios.

The new assessment of accident consequences will consider (1) enhancements in plant design, operation, inspection, maintenance, and accident management; (2) security-related enhancements made by plant owners or implemented in response to requirements that the NRC has issued since the terrorist attacks on September 11, 2001; and (3) improvements in calculation methods for accident progression and consequences analyses. In order to reflect the plant enhancements, including EP modifications, made in response to risk assessments and security requirements, this new consequence assessment will require a coordinated effort by the Offices of Nuclear Regulatory Research (RES), Nuclear Reactor Regulation (NRR), and Nuclear Security and Incident Response (NSIR), to gather supporting plant-specific information from licensees. In some areas, such as EP, the assessment will require site-specific information, for which NSIR will have the lead responsibility. (Some sites have updated their evacuation time estimates, while other sites may rely on earlier conservative estimates that may need to be updated.) In other assessment areas, primarily those related to scenario development and systems response, sufficient information will be needed to ensure the applicability of the scenarios for each plant. NRR and NSIR will have the primary lead for providing this information, some of which may be solicited from licensees. RES will have the role of identifying specific information needs to support the accident consequence assessment.

In conducting this assessment, the staff will integrate existing state-of-the-art analytical tools, which the NRC is currently using, into a coherent methodology that can be used to better understand realistic best-estimate radiological dose consequences (including early and latent fatalities and land contamination) of any severe accident sequence selected for study. Generic MELCOR calculations, based on major plant classes of boiling- and pressurized-water reactors (BWRs and PWRs), will be used to determine the time to fuel failure, as well as the magnitude and timing of environmental fission product releases. We will then use these results to perform site-specific consequence evaluations for each risk-significant accident sequence identified for consideration.

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[1] The EPRI RAMCAP study developed generic fuel damage scenario classes for boiling- and pressurized-water reactors (BWRs and PWRs, respectively). Those scenario classes encompass loss-of-coolant accidents (LOCAs) without reactor pressure vessel (RPV) injection, small LOCAs without RPV injection, short-term station blackout, long-term station blackout, loss of offsite power, loss of ultimate heat sink, and loss of spent fuel pool integrity. Another class is designated for any plant-unique cases. These classes are generally consistent with other studies that identify risk-significant accident scenarios, such as plant-specific IPEs, as summarized in NUREG-1560, "Individual Plant Examination Program: Perspective on Reactor Safety and Plant Performance," dated December 1997.

The following table summarizes the overall process for developing state-of-the-art consequence analysis. To augment that summary, the next section presents a background discussion that compares the results of Sandia's 1982 Siting Study with those of recent security analyses. The remainder of this paper presents a detailed plan with milestones and costs for conducting the study.

Overall Process for Developing State-of-the-Art Consequence Analysis

Process Step	Essential Consideration	Related Risk Reduction
Identify representative severe accident scenarios: Selection of scenarios based on risk insights, informed by EPRI RAMCAP generic fuel damage scenario classes and identification of risk-significance derived from site-specific IPE studies (NUREG-1560). Anticipate general scenarios consistent with selected RAMCAP general fuel damage scenario classes.	**Selection of risk-significant scenarios:** • radiological release frequency of greater than or equal to 10^{-6} per reactor year • must reflect current plant design, layout, operation, accident management, and security enhancements Scenarios would be prioritized according to risk-significance.	Identify measures to reduce the likelihood of interfacing systems LOCA if radiological release frequency is $\geq 10^{-6}$/RY and is a significant risk contributor. Consider the contribution of security assessment plant improvements to reduce the likelihood of core damage.
Evaluate representative source terms: Accident progression, as well as the magnitude and form of fission product release	Perform MELCOR calculations (use existing calculations where appropriate) that employ best-practice calculation techniques, realistic models, and phenomenology to estimate the following: <u>Accident Timing</u> • Time to uncover the core • Time to core damage • Time to vessel failure • Time to containment failure <u>Environmental Source Term</u> • Large early (prior to completion of evacuation) • Large late (after completion of evacuation) • Moderate (magnitude reduced by fission product retention) • Small (magnitude significantly reduced)	Identify measures to reduce the event likelihood, as well as measures to delay core damage or containment failure (where not evaluated in previous studies). In particular, identify measures to reduce the probability of bypass scenarios (e.g., interfacing systems LOCA), if radiological release frequency is $\geq 10^{-6}$/RY and is a significant risk contributor. Identify additional measures to enhance retention and scrubbing of fission products and estimate consequence reduction.

Process Step	Essential Consideration	Related Risk Reduction
Estimate consequences	Perform site-specific MACCS2 calculations (use existing calculations where appropriate) that employ realistic models and phenomenology, best-practice calculation techniques, and site-specific EP to estimate: • Early fatalities (Phase I) • Latent cancer fatalities (Phase II) • Environmental impact (Phase III) Determine latent health effects for a spectrum of assumed thresholds.	Identify EP features to ameliorate consequences.

Background

The most recent security assessments indicate much smaller potential offsite consequences from severe accidents (e.g., fuel melt) than those often portrayed in earlier probabilistic risk assessments (PRAs) and consequence studies (e.g., NUREG/CR-2239, commonly referred to as "Sandia's 1982 Siting Study"). The smaller predicted consequences in the security assessments (compared to Sandia's 1982 Siting Study) are primarily attributable to the following factors:

- more realistic accident progression and consequence modeling
- more realistic emergency preparedness assumptions
- differences in the spectrum of accidents considered

The commonly cited consequences from the 1982 Siting Study relate to a very severe scenario designated as "siting source term one," or SST1. That report presents predicted offsite consequences for a distribution of weather conditions, including both mean values and low-probability weather conditions. Advocacy groups (such as Riverkeeper) usually cite the 99.5th or 99.9th percentile values. The use of these values corresponds to orders of magnitude increases in calculated early fatalities, often attributable to the large predicted effect of rainfall occurring at the worst possible time and washing out radionuclides over population centers. The obvious argument against using such low-probability outcomes is that using the 99.9th percentile outcome for an event that has a probability of 10^{-6} per reactor year effectively transforms it into a likelihood of 10^{-9} per reactor year. We should also reexamine the technical rigor and appropriateness of the analysis of washing out the radionuclides and resultant population exposure.

The following table compares the 1982 Siting Study consequences with those predicted in the recent security assessment of the reference BWR plant.

BWR Results

	1982 Siting Study, SST1 scenario (mean values)	1982 Siting study, SST1 scenario (99.9th percentile)	Security assessment — bypass type scenario (mean value)	Security assessment — SBO type scenario (mean value)
Early fatalities	92	~15,000	0	0
Latent cancer fatalities	2662	>30,000	2,000 – 14,000 (depending on threshold)	70 – 5,000 (depending on threshold)

The differences between the predicted results are significant because of the factors cited above. The BWR bypass type scenario does not exhibit a great deal of difference in the severity of the radiological release. However, the 1982 Siting Study used a generic treatment of EP, assuming evacuation delay times of 1, 3, and 5 hours, with 30%, 40% and 30% probabilities, respectively. By contrast, the site-specific EP evaluation for the reference BWR determined that an evacuation delay time of 45 minutes was appropriate for that site. Thus, even the fast bypass type scenario would be mitigated by EP for the reference BWR. Moreover, the more slowly developing SBO type scenario showed a substantial margin in time available for EP. Similarly, for the more probable SBO-type scenario, the magnitude and timing of the radiological release is much less severe for the reference BWR than the Siting Study SST1 source term. In addition, the release timing for the SBO scenario is notably longer than for the SST1 source term. (SST1 release is assumed to begin in 1.5 hours.)

For PWR plants, the 1982 Siting Study is similarly dominated by the SST1 source term. In the security assessment, there was no fast bypass type scenario (comparable to that for the BWR), which might also have been comparable to an SST1 type release. In the security assessment, there was only a slowly developing SBO type scenario, with releases occurring much later than for the BWR. Containment failure and offsite releases did not occur for days. The table below compares predicted offsite consequences for the Siting Study with those predicted in the recent security assessment of the reference PWR plant.

PWR Results

	1982 Siting Study, SST1, (mean value)	1982 Siting Study, SST1, (99.9th percentile)	Security assessment — SBO type scenario (mean value)
Early fatalities	45	~20,000	0
Latent cancer fatalities	1,200	>20,000	0 – 70 (depending on threshold)

Study Conduct

The proposed study will make full use of and capitalize on previous studies, take advantage of progress made in security-related work, and incorporate insights gained from recent safety-related studies.

With regard to scenario selection, we will review existing PRAs, IPEs, SPAR, and RAMCAP studies to identify representative severe accident scenarios for each type of plant (i.e., PWRs and BWRs). The scenario classes may encompass LOCAs without RPV injection, small LOCAs without RPV injection, short-term SBO, long-term SBO, loss of offsite power, and loss of ultimate heat sink. For any scenario with a radiological release frequency of less than 10^{-6} per reactor year, no representative source term or consequence calculations will be performed.

In conducting this study, we will also factor in insights gained from extensive NRC research programs on containment performance and severe accident phenomenology. For example, we will incorporate insights from Sandia's testing of steel and pre-stressed concrete containment models, effect of containment leakage versus catastrophic containment failure; effects of measures to reduce the likelihood of liner melt-through of BWR Mark I containment, reduced likelihood of failure by direct containment heating, and so forth.

In conducting this study, we will also assign a representative source term to each scenario type. These source terms will be based on realistic MELCOR calculations, which account for the different depletion mechanisms in the reactor coolant and containment systems. Where appropriate, we will use existing MELCOR calculations, supplemented with additional calculations. Similarly, where appropriate, MELCOR analyses will parametrically credit readily available mitigation measures, as well as additional (i.e., beyond readily available measures) active systems (e.g., external spray, aerosol scavenging, or foam) to mitigate offsite releases.

Finally, with regard to the consequence analysis, we will use the MACCS2 code to generate site-specific consequences that account for weather conditions, population distribution/density, and EP (sheltering, relocation, and evacuation). We will report mean values for current plant configurations and selected scenarios that credit additional mitigative systems. The study will identify the value of these mechanisms that should inform decisions regarding future research to optimize system effectiveness through analytical studies and experimental verification. Phase I of the study will focus on early fatalities, while Phase II will address latent cancer fatalities, and Phase III will address land contamination.

Analysis Methods

Identify Representative Severe Accident Scenarios

Scenarios will be selected, for general classes of reactors, based on their contribution to risk. Each selected scenario will be reviewed to ensure that it would still be considered risk-significant, given recent plant modifications, current estimates of failure probabilities (e.g., pipe break frequency, emergency diesel reliability), and appropriate credit for emergency procedures and accident management strategies (including post-9/11 security enhancements). Scenarios that are determined to have an overall radiological release frequency less than 10^{-6}/yr will be screened from consideration. Scenarios with a frequency that exceeds 10^{-6}/yr will be ranked according to their risk-significance to ensure that those with the greatest contribution to overall risk are evaluated first.

Evaluate Representative Source Terms

MELCOR will be used to perform integral severe accident (in-vessel and ex-vessel progression, containment response, and fission product release, transport, and retention) analyses to supplement existing analyses. MELCOR input decks exist for the plants that were evaluated for security work, but can be modified for this study at reasonable cost. A Mark II containment model will be developed as part of this study. The models will need to be assessed to ensure that severe accident management strategies are appropriately credited.

MELCOR calculations will reflect current best practice modeling techniques and assumptions that represent the best estimate of accident progression for a given scenario (with no intentional conservatism built into the analyses). Prior to the start of the plant analyses, any potential substantive improvements to the modeling, which can implemented in a timely cost-effective manner, will be identified and implemented. Phenomenological and sequence uncertainties, if considered important to overall risk (i.e., different assumptions might result in different accident progressions), will be treated. Calculations will consider research insights that would affect overall accident progression timing and source term magnitude (e.g., experimental insights regarding fission product release from Phébus and VERCORS programs; containment failure pressure, size, and location insights from the containment performance testing program; etc.). MELCOR calculations will also be performed to confirm the effectiveness of identified readily available additional measures to prevent core damage or containment failure, as well as to demonstrate the value of proposed beyond readily available measures (discussed later in this paper).

Estimate Consequences

Site-specific consequence calculations will be performed, with emphasis on realism that avoids needless conservatism. Site-specific data will be obtained for population distribution, meteorology, and parameters for modeling emergency response. The population distributions are readily available from the 2000 census data, using an existing NRC code that reads the Census Bureau files. Each plant has an evacuation time estimate for its 10-mile emergency planning zone (EPZ), and ad hoc emergency response will be considered beyond the EPZ. Licensees are required to measure certain meteorological parameters, and those files will be requested. If site-specific meteorology data are not readily available or retrievable in a timely manner, data from the nearest National Oceanic and Atmospheric Administration (NOAA) meteorology station may be substituted. For those States that have obtained potassium iodide (KI) for their residents, ingestion of KI will be included in the calculations. Consequence calculations will also be performed to confirm the adequacy of identified EP enhancements.

Several metrics will be considered for inclusion in the study, and a phased approach will be used in reporting the consequence results:

- Phase I – Early fatalities
- Phase II – Latent cancer fatalities
- Phase III – Land contamination and economic consequences

A new editing option has been added to MACCS2 to display the amount of land contaminated above a given value. Various options for the threshold "contaminated" value will also be investigated to provide perspective. It is anticipated that Phase III analyses will require Commission guidance on policy and criteria for responses to severe events. Prior to beginning site analyses, the MACCS code will be reviewed to evaluate any substantive, cost-effective modeling improvements which can be implemented in a timely manner, and those improvements will be implemented.

Identification and Valuation of Beyond Readily Available Mitigative Measures

Based on the consequence analyses described above, the staff will identify mitigative measures that have the potential to significantly reduce risk or conditional consequences. Slowly breaking scenarios provide opportunities for mitigation. In contrast, fast breaking scenarios may be more effectively mitigated by preventive measures. Measures that may limit the quantity of released aerosols that are transported offsite will be considered. (These are briefly discussed below.)

Preliminary investigation of active mitigation concepts to limit offsite releases following containment failure has been performed as a follow-on to the pilot plant security assessments. Early evaluations have identified a set of fundamental characteristics that any mitigation concept must have to be effective:

- The mitigation option must have the ability to capture airborne radioactive aerosols as, or just after, they are released from a nuclear plant building (the release may not always come from the containment) and then retain the captured material on site.
- The mitigation option must be deployable independent of any onsite power source.
- The mitigation option must be operable in a high-radiation environment, as the initial release of noble gases could generate dose rates as high as 100s of rad/hr within 50 meters of the release point.
- The mitigation option must be effective in the outside ambient environment over a wide range of environmental conditions (e.g., wind speed, temperature, humidity).

These characteristics need to be confirmed to develop an optimal engineered solution. Because of the required investment for experimental verification of system efficacy and engineered design, it is prudent to perform initial analytical studies to evaluate the potential reduction in offsite release offered by proposed systems. Three promising candidates for active mitigation systems that could be deployed separately or in combination are use of fog or water sprays, use of foam technologies, and use of enhanced agglomeration agents, as follows:

- *Mitigation through Use of Fog or Water Sprays*

  Water sprays would have the potential to scrub aerosol particles and condensable vapors from the plume in the vicinity of the release point, and would suppress the thermal buoyancy of the plume, thereby making the release easier to contain. Preliminary analyses show that control of an aerosol and vapor plume associated with a severe accident could be accomplished by spraying water at a rate that could be supplied by a few fire trucks. Assuming that the optimal water droplet size distribution and aerial flux could be achieved, this mitigation approach is appealing because of its simplicity and dependence upon readily available resources (i.e., water, pumps, and personnel trained in fire suppression).

- *Mitigation through Use of Foam Technologies*

  Foam technologies have the potential to mitigate radioactive source term effects in a manner similar to sprays, but retain fission products within an onsite structure. It may be possible to deeply flood large buildings (from which a source term is emerging) with foams that could both suppress fires and provide fission product scrubbing and retention.

- *Mitigation through Use of Enhanced Agglomeration Agents*

  The size distribution of the radioactive aerosol particles emerging from a damaged reactor system will likely be in the range of 0.5 to 5 micrometer, based on experimental studies and aerosol mechanics analyses. This size range is not optimal for the most effective water spray scrubbing, and considerable improvement in spray effectiveness could be gained if the target aerosol could be manipulated to larger sizes (in the range of 10 to 20 micrometers). This might be accomplished using enhanced agglomeration agents with larger inert particles. Potential candidates include fog mists or dense man-made smokes.

The assessment of beyond readily available measures for this project will consist of two parts. First, MELCOR scoping calculations will be performed to assess potential benefits and preliminary design criteria for beyond readily available mitigation measures. These calculations will adapt existing MELCOR models to estimate the source term reduction offered by active mitigation systems for selected scenarios. In pursuit of these goals, some additional standalone analyses will be performed to provide the basis for spray and foam mitigation effects to be factored into the MELCOR and MACCS2 assessments. Results of these MELCOR calculations will form the basis for identifying measures that show promise for significant source term reduction and that should be examined experimentally. Descriptions and cost estimates for experiments that would demonstrate the efficacy of any proposed measure would be produced as a part of these preliminary calculations. Second, these same mitigation measures would later be incorporated into MELCOR (and then MACCS2) calculations for select scenarios that are being evaluated as part of the effort for developing state-of-the-art consequence analysis. This will demonstrate the potential consequence reduction for risk-significant scenarios identified early in this study for a few high-population sites.

Development of Faster than Real-Time Computer-Based Decision-Making Tool

The staff will develop an integrated faster than real-time, computer-based decision-making tool, which can be used to enhance NRC and Federal responses to events of national significance. Toward that end, the staff will obtain input from computer code modeling efforts conducted by IEMA.

As a separate, but coordinated and parallel activity, the staff proposes to bring a best-estimate approach, founded on integrated modeling, to incident response and management. As a result of developing state-of-the-art consequence analyses described in this paper, we will have developed a large collection of accident signatures and plant models for the U.S. fleet. As part of the task to develop real-time decision-making capability, this accident signature database would be part of a developed software package that also includes MELCOR models or sub-models to provide the capability to interactively assess variations in those sequences to represent departures from the pre-calculated accident progression signatures. The capability to analyze variations in accident progression signatures would also include the ability to model the effects of intervention or mitigation. As part of this same decision-making capability, the code would model offsite health effects and land contamination using real event weather and site characteristics (including EP and variations). Both the MACCS code and other offsite models (e.g., RASCAL) would be considered for this offsite consequence modeling. (These codes are already quite fast-running.) As a tool for emergency management, this code will be faster than real-time and would be developed for use by NRC personnel in the Headquarters Operations Center. Training would also be provided for NRC personnel.

Another option would be to produce a simplified plant model capable of running faster than real-time. Presently, it is not clear how much runtime acceleration is possible through nodalization simplifications, etc., that preserve the essential timing accuracy of the predictions. Since significantly faster-than-real-time performance is desired, significant simplification would be required. This option should be investigated further as it allows use of many other MELCOR features, such as mitigative actions of sprays, regained plant systems, and so forth.

In addition to a faster-than-real-time capability to predict realistic source term signatures, an ability to evaluate the implications of this developing source term on emergency actions is also needed. This can be assisted by integrating the source term information with an atmospheric transport analysis tool [such as MACCS, RASCAL, or possibly one of the transport codes developed by Lawrence Livermore National Laboratory (LLNL)]. Desired features of this tool will be to accept time-dependent source term information from MELCOR and then make use of local topological information and weather data to predict likely local dose rate and land contamination information. This would require a puff release model capable of changing atmospheric transport direction and tendencies owing to terrain and wind changes. Attractive features in MACCS include dose assessment and land contamination prediction, whereas RASCAL includes a puff model for atmospheric transport. By contrast, the LLNL code suite has considerable national recognition for state-of-the-art capabilities. The feasibility of employing such codes should be assessed.

Project Organization

- The Office of Nuclear Regulatory Research (RES) will have overall responsibility for project management and technical direction. The RES staff will be involved in all aspects of the effort and will be responsible for issuing the report.

- The Office of Nuclear Reactor Regulation (NRR) will be consulted in event selection and probabilistic quantification, and will provide the meteorology files for each site, along with certain plant data that may be needed to construct realistic MELCOR and MACCS2 input decks. NRR will also provide plant-specific information needed to confirm the appropriateness of scenario and system modeling.

- Scenario selection will be identified by the staff (RES and NRR) and discussed with the Advisory Committee on Reactor Safeguards (ACRS) and, if needed, a group of experts, to ensure soundness of the process.

- Contractor expertise will be required. Contractor staff will perform the MELCOR analyses for the various plant types, as well as the bulk of the MACCS2 consequence analyses. RES will conduct selected in-house MACCS2 calculations.

- The Office of Nuclear Security and Incident Response (NSIR) will provide the emergency response parameters used in the consequence analyses. Different values will be required for each site. NSIR will also provide generic and plant-specific information on security enhancements that relate to plant response for selected scenarios.

- Senior management oversight of the project will be provided through an agency-level steering committee. The steering committee will be briefed on progress, key technical and programmatic issues in directing project completion and success, and results as they become available. Steering committee review will be conducted every 6 months.

Commission Interactions

The staff will keep the Commission informed annually and provide briefings to the Commissioners' Technical Assistants every 6 months.

The staff will also provide, for Commission consideration, options on the extent to which land contamination and offsite economic consequences should be addressed in developing recommendations for mitigative measures, and how best to achieve that objective.

Cost[2]

The staff estimates that the contractor level of effort will total $7.45M spread over 3 years, with a completion date of March 2009, as follows:

Developing State-of-the-Art consequences		+	**Real-time decision-making tool**	
FY 2006	$2,550K		FY 2006	$500K
FY 2007	$2,200K		FY 2007	$500K
FY 2008	$1,200K		FY 2008	$500K

The consequence calculations for high-population sites will be completed by October 2006. The additional experimental verification of mitigative system efficacy, if approved, will involve efforts to develop performance criteria for beyond readily available mitigation measures.

Budget Breakdown of Developing State-of-the-Art Consequences	**Staff Months**
Develop MELCOR plant models	22
Develop consistent MELCOR standard practice assumptions	2
Perform MELCOR Calculations	80
Develop updated EP modeling assumptions	20
MACCS2 analyses	33
Develop performance criteria for beyond readily available mitigation measures	3
MELCOR calculations to demonstrate active mitigation value	3
Reporting	29
Total Project Duration	**192**

The level of effort assumed for MELCOR analyses assumes roughly four base scenarios per reactor class, additional parametric calculations to treat important phenomenological uncertainties, and calculations to demonstrate the impact of identified readily available mitigation measures.

Experimental work to demonstrate the efficacy of mitigative strategies would follow if deemed valuable based on the analytical work performed in this project. Schedule and cost estimates would be produced as part of the effort during FY 2006.

The staff also anticipates that the proposed plan will require a total of approximately 12 staff full-time equivalents (FTE) over the 3-year period of the program for project management and coordination, technical direction and support, and review. This estimate includes 9 FTE from RES, 1 FTE from NRR, and 2 FTE from NSIR.

2 The cost estimate shown is for contractor support and does not include the NRC staff FTE.

Schedule

Expected Start Date:	Upon Approval of the Plan
Expected Completion Dates[3]:	
Performance Criteria for Beyond Readily Available Measures	05/2006
High-Population BWR Mark I Sites	12/2006
High-Population PWR Westinghouse 4-Loop, Large Dry Sites	12/2006
Calculations to Demonstrate Active Mitigation Value	10/2006
Project Completion	3/2009

Figures 1 and 2 provide additional schedule details.

Deliverables

1. A "living" collection of accumulated knowledge of physical experiment results, analytical studies, and computer codes that provide the bases for determining the most influential variables and assumptions, and that informs the nature and extent of the associated uncertainties.

2. A research plan, which identifies recommendations for cost-beneficial experimental or other research that should be undertaken to improve the accuracy of the analytical tools, or reduce uncertainties in key parameters that drive the associated risk. The plan should also estimate the completion of approved research, and incorporation of results into applicable codes and analysis methods.

3. A publicly available appendix to NUREG for traditional reactor safety risk analysis.

4. Both public and non-public versions of a supplement to the NUREG for security-related scenarios.

5. A library of pre-calculated high-fidelity analyses, using the MELCOR code, which cover the range of scenarios possible for plant designs in use in the United States.

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3 Completion dates assume a project start date no later than January 3, 2006.

Figure 1. Schedule for Rebaselining of Reactor Risk and Off-site Release Mitigation Analyses.

Plant Class	2006											
	J	F	M	A	M	J	J	A	S	O	N	D
BWR, Mark I												
BWR, Mark II												
BWR, Mark III												
PWR, B&W												
PWR, CE												
PWR, W 4-Loop, Large Dry												
PWR, W 3-Loop, Subatm.												
PWR, W 4-Loop, Ice Cond.												
Off-site Release Mitigation												

Plant Class	2007											
	J	F	M	A	M	J	J	A	S	O	N	D
BWR, Mark I												
BWR, Mark II												
BWR, Mark III												
PWR, B&W												
PWR, CE												
PWR, W 4-Loop, Large Dry												
PWR, W 3-Loop, Subatm.												
PWR, W 4-Loop, Ice Cond.												
Off-site Release Mitigation												

Plant Class	2008											
	J	F	M	A	M	J	J	A	S	O	N	D
BWR, Mark I												
BWR, Mark II												
BWR, Mark III												
PWR, B&W												
PWR, CE												
PWR, W 4-Loop, Large Dry												
PWR, W 3-Loop, Subatm.												
PWR, W 4-Loop, Ice Cond.												
Off-site Release Mitigation												

Legend

Best Practices Review (MELCOR and MACCS)

MELCOR Model Development / Upgrades, MACCS Data Collection

MELCOR Calculations, MACCS Deck Development

MACCS Calculations, High Population Sites

MACCS Calculations, Remaining Sites (All Sites)

Develop Performance Criteria for Beyond Readily Available Mitigation Measures

Mitigation Calculations

Reporting

Figure 1. Schedule for Rebaselining of Reactor Risk and Off-site Release Mitigation Analyses (continued).

Plant Class	2009											
	J	F	M	A	M	J	J	A	S	O	N	D
BWR, Mark I												
BWR, Mark II												
BWR, Mark III												
PWR, B&W							Project Completed					
PWR, CE												
PWR, W 4-Loop, Large Dry												
PWR, W 3-Loop, Subatm.												
PWR, W 4-Loop, Ice Cond.												
Off-site Release Mitigation												

Legend

Best Practices Review (MELCOR and MACCS)

MELCOR Model Development / Upgrades, MACCS Data Collection

MELCOR Calculations, MACCS Deck Development

MACCS Calculations, High Population Sites

MACCS Calculations, Remaining Sites (All Sites)

Develop Performance Criteria for Beyond Readily Available Mitigation Measures

Mitigation Calculations

Reporting

Figure 2. Schedule for Development of Real-time Incident Response and Decision-making Tool.

Activity	2006											
	J	F	M	A	M	J	J	A	S	O	N	D
Software Rqmts Document												
Dispersion Model Recomm.												
GUI Design												
GUI Development/Refinement												
Meteorology Data Engine												
Integrate Accident Database												
Integrate Simplified MELCOR												
Documentation												
NRC Staff Training												

Activity	2007											
	J	F	M	A	M	J	J	A	S	O	N	D
Software Rqmts Document												
Dispersion Model Recomm.												
GUI Design												
GUI Development/Refinement												
Meteorology Data Engine												
Integrate Accident Database												
Integrate Simplified MELCOR												
Documentation												
NRC Staff Training												

Activity	2008											
	J	F	M	A	M	J	J	A	S	O	N	D
Software Rqmts Document												
Dispersion Model Recomm.												
GUI Design												
GUI Development/Refinement												
Meteorology Data Engine												
Integrate Accident Database												
Integrate Simplified MELCOR												
Documentation												
NRC Staff Training												

April 14, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

FROM: Kenneth R. Hart, Acting Secretary /RA/

SUBJECT: STAFF REQUIREMENTS - SECY-05-0233 - PLAN FOR DEVELOPING STATE-OF-THE ART REACTOR CONSEQUENCE ANALYSES

The Commission has approved the staff's plan to (1) evaluate and update, as appropriate, analytical methods and models for realistic evaluation of severe accident progression and offsite consequences; (2) develop state-of-the-art reactor consequence assessments; and (3) develop an integrated, predictive, computer-based tool to assist decision-making in the event of a severe reactor accident.

The staff shall ensure that the updated study results include a written discussion (non-public if necessary) of the extent to which security-related initiating event scenarios are addressed by the release groups into which the spectrum of accident scenarios are binned and the completed and ongoing security assessments (i.e., phases 1, 2, and 3).

The staff should seek Commission approval prior to conducting analyses for security related events that are not captured by the spectrum of scenarios adopted for the consequence analyses. Such security related events may have been encompassed by the work undertaken in response to the events of September 11, 2001. Therefore, the staff should provide a summary of the benefits that would be gained from conducting this additional work in view of the security related analyses that have been completed or are under way. The Commission supports development of a non-public version of the study for security related events if analyses for such events are conducted.

The staff should complete this work through a coordinated effort by the Offices of Nuclear Regulatory Research (RES), Nuclear Reactor Regulation (NRR), and Nuclear Security and Incident Response (NSIR).

The staff's proposal to examine significant radiological release scenarios having estimated likelihoods of one in a million or greater per year, is an appropriate initial focus. This initial set of analyses should focus attention on the scenarios of greatest interest and provide useful insights into the effectiveness of current and postulated mitigation strategies. To the extent practicable, all new analyses should account for enhancements implemented by licensees in the areas of safety and security and should use state-of-the-art analytical tools for accident progression and consequence analyses. The staff should keep the Commission up-to-date on the results and status of the site-specific consequence analysis.

The staff should use the improved understanding of source terms and severe accident phenomenology (e.g., containment failure modes, time of release, release duration, inventory release fractions), and credit the use of Severe Accident Management Guidelines (SAMGs) and other new procedures, such as mitigative measures resulting from B.5.b and other like programs, that were not in place when the earlier study was performed. The staff should also utilize updated and realistic plant specific information for other variables such as surrounding population, meteorology, and evacuation assumptions.

The staff should present its updated results using risk communication techniques to achieve an informed public understanding of the extent and value of defense-in-depth features including current mitigative strategies, and of the important analytical assumptions. In presenting these results, the staff needs to develop substantial improvements to the communication and presentation techniques that were used previously in NUREG/CR-2239 (1982 siting study); this includes a discussion of the differences between the state-of-the-art analysis and that reported in the NUREG/CR.

In the paper, the staff presents some of the results of its recent analyses as examples, but these have not fully benefitted from the staff's proposed new methodology, and therefore this paper should not be made public at this time. The results of the proposed analyses, and their underlying bases, should be made public as an important objective of this initiative. To better communicate the results to our stakeholders, the staff should properly characterize the uncertainties in the results and identify the significant influential inputs and assumptions.

In applying a screening radiological release frequency of 10^{-6} per reactor year (i.e., to analyze only those scenarios that have a release frequency of greater than 1 in a million), the staff should be careful to define release groupings such that release characteristics are representative of scenarios binned into those groups. However, where possible, the groups should also be sufficiently broad to be able to include the potentially risk-significant but lower frequency scenarios (for example, the interfacing systems LOCA scenarios that bypass the containment).

Potential offsite health effects are very dependent on the evacuation model used.. Realistic site-specific evacuation scenarios should be incorporated and basis for the inputs on delay times, evacuation speeds, and fractions of non-evacuating population should be discussed.

As part of implementation of the plan, the staff should work with the ACRS on technical issues such as identification of accident scenarios to be evaluated, evaluation of source terms, credit for operator actions or plant mitigation systems, modeling of emergency preparedness, modeling of offsite consequences, and definition and characterization of analysis uncertainty.

In performing the consequence analysis, the staff should rely on currently available methods and models. Tasks such as the experimental validation of beyond readily available mitigative measures should be discussed with the Commission after results from the base case consequence analysis become available.

The scope of these analyses may include mitigation strategies that are required under section B.5.b of the Commission's February 25, 2002 Order to power plant licensees or any superseding regulation, and may further include additional strategies to which licensees have committed as a result of the previously completed and ongoing security assessments (i.e., phases 1, 2, and 3). The staff shall evaluate other significant and appropriate mitigation strategies for radiological consequences in a separate study, starting with scoping evaluations, and should keep the Commission fully informed of its progress with these evaluations during the periodic security briefings.

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
OGC
CFO
OCA
OPA

TAB 8

538TH ACRS MEETING
DECEMBER 8 , 2006
ROCKVILLE, MARYLAND

Proposed Revisions to Regulatory Guides and Standard Review Plan Sections

8:35 am - 9:30 am   Discussion Lead by Otto Maynard

Attachments:

1. Table Showing Status of ACRS Review of High Priority Regulatory Guides
2. Table Showing Status of ACRS Review of High Priority Standard Review Plan Sections

Cognizant ACRS Staff Engineer: David Fischer

NRC Staff Available to Answer Questions: Steve Koenick, NRR (for SRP Sections)
John Ridgely, RES (for Reg Guides)

Revised: November 8, 2006

STATUS
ACRS REVIEW OF HIGH PRIORITY REGULATORY GUIDES

RG No.	Regulatory Guide Title	Mbr/Eng	Status
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident (See SRP 6.2.5)	WJS/EAT	Letter 11/??/06
1.9 DG-1172	Application and Testing of Safety Related Diesel Generators in Nuclear Power Plants	JDS/MAJ	Don't Review Larkinsgram 10/16/06
1.13 DG-1162	Spent Fuel Storage Facility Design Basis (See SRP 9.1.1 and 9.1.2)	DAP/HPN	Don't Review Larkinsgram 10/16/06
1.20 DG-1163	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	JSA/MB	Don't Review Larkinsgram 10/16/06
1.23 DG-1164	Onsite Meteorological Programs (See SRP 2.3.3)	TSK/DCF	Don't Review Larkinsgram 9/13/06
1.26 DG-1152	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (See SRP 3.2.2)	JSA/MB	Don't Review Larkinsgram 10/16/06
1.29 DG-1156	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	GEA/HPN	Don't Review Larkinsgram 10/16/06

RG No.	Regulatory Guide Title	Mbr/Eng	Status
1.37 DG-1165	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	OLM/MAJ	Don't Review Larkinsgram 10/16/06
1.57 DG-1158	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	JSA/MB	Don't Review Larkinsgram 10/16/06
1.61 DG-1157	Damping Values for Seismic Design of Nuclear Power Plants	DAP/HPN	Don't Review Larkinsgram 11/3/06
1.68 DG-1166	Initial Test Programs for Water-Cooled Nuclear Power Plants	MVB/MB	Don't Review Larkinsgram 10/16/06
1.71 DG-1167	Welder Qualification for Areas of Limited Accessibility	JSA/MB	Don't Review Larkinsgram 10/16/06
1.76 DG-1143	Design Basis Tornado for Nuclear Power Plants (See SRP 2.3.1 and 3.5.1.4)	MC/MAJ	Don't Review Larkinsgram 9/13/06
1.93 DG-1153	Availability of Electric Power Sources	JDS/MAJ	Don't Review Larkinsgram 10/16/06
1.112 DG-1160	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors	JDS/MAJ	Referred to ACNW Larkinsgram 10/16/06

RG No.	Regulatory Guide Title	Mbr/Eng	Status
1.124 DG-1168	Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports	WJS/CXS	Don't Review Larkinsgram 10/16/06
1.128 DG-1154	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants	OLM/RC	Don't Review Larkinsgram 10/16/06
1.129 DG-1155	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants	OLM/RC	Don't Review Larkinsgram 10/16/06
1.130 DG-1169	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports	JSA/CXS	Don't Review Larkinsgram 10/16/06
1.136 DG-1159	Materials, Construction, and Testing of Concrete Containments (DG-1159)	WJS/CXS	Don't Review Larkinsgram 11/3/06
1.189 DG-1170	Fire Protection for Operating Nuclear Power Plants (See SRP 9.5.1)	JDS/MAJ	Reviewed in Nov.
1.196 DG-1171	Control Room Habitability at Light-Water Nuclear Power Reactors	DAP/EAT	Don't Review Larkinsgram 10/16/06
1.200 DG-1161	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	GEA/EAT	Letter 10/23/06
DG-1142	Guidelines for Environmental Qualification of Safety Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants	SAK/EAT	Don't Review Larkinsgram 10/16/06

RG No.	Regulatory Guide Title	Mbr/Eng	Status
DG-1144	Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light Reactor Water Environment for New Reactors * (See SRP 3.12)	JSA/CXS	Review in Dec.
DG-1145	Combined License Applications for Nuclear Power Plants (LWR Edition) *	TSK/DCF	Review in Dec.
DG-1146	Seismic Sources and Safe Shutdown Earthquake Ground Motion (See SRP 2.5.2)	DAP/HPN	Don't Review Larkinsgram 11/3/06
4.15	Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment	OLM/DCF	Referred to ACNW Larkinsgram 10/16/06

STATUS
ACRS REVIEW OF HIGH PRIORITY STANDARD REVIEW PLAN SECTIONS

SRP SECTION	SRP Section Title	Mbr/Eng	Received	Status
2.3.1	Regional Climatology (See RG 1.76)	MC/MAJ		
2.3.3	Onsite Meteorological Measurements (See RG 1.23)	TSK/DCF		
2.4.6	Probable Maximum Tsunami Flooding	DAP/CGH		
2.5.2	Vibratory Ground Motion (See DG-1146)	DAP/HPN		
3.2.1	Seismic Classification	GEA/HPN	draft 10/5/06 formal 11/6/06	Don't Review
3.2.2	System Quality Group Classification	JSA/MB	draft 10/5/06 formal 11/6/06	Don't Review (verify with JSA)
3.12	ASME Code Class 1, 2, and 3 Piping Systems and Associated Supports Design [new] (See DG-1144)	JSA/CXS	draft 10/17/06	Don't Review
3.13	Threaded Fasteners - ASME Code Class 1, 2, and 3	WJS/MB	draft 10/17/06 formal 11/6/06	Don't Review
4.2	Fuel System Design	JSA/RC		

5.4.8	Reactor Water Cleanup System (BWR)	JDS/MAJ	draft 10/23/06	TBD: Don't Review (tent)
6.2.5	Combustible Gas Control Containment	WJS/EAT	10/2/06	Letter 11/??/06
7.8	Diverse I&C Systems	GEA/EAT		
BTP 7-19	Guidance for the Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems	GEA/EAT		
9.1.1	New Fuel Storage	JSA/RC		
9.1.2	Spent Fuel Storage	TSK/HPN		
9.1.3	Spent Fuel Pool Cooling and Cleanup System	DAP/HPN	8/25/06	Don't Review Larkinsgram 11/6/06
9.5.1	Fire Protection Program (See RG 1.189)	JDS/MAJ	draft 10/25/06 formal 11/6/06	Reviewed Nov. Revisit Dec.
10.3.6	Steam and Feedwater System Materials	JSA/CXS	8/24/06	Don't Review Larkinsgram 11/6/06
11.2	Liquid Waste Management Systems	JDS//MB		
11.3	Gaseous Waste Management Systems	JDS/MB		
11.4	Solid Waste Management Systems	JDS/MB		
12.3 -12.4	Radiation Protection Design Features	DAP/CGH		
13.3	Emergency Planning	MC/MB	9/19/06	Review in Dec.
15.0	Accident Analysis - Introduction	SB/RC		
15.9	BWR Core Stability [new]	SB/RC		

17.4	Reliability Assurance Program [new]	GEA/EAT	10/31/06	TBD
17.5	Quality Assurance	TSK/DCF	9/22/06	Don't Review Larkinsgram 11/6/06