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

September 11, 2008

AGENDA
556th ACRS MEETING
OCTOBER 2-4, 2008

THURSDAY, OCTOBER 2, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - 10:00 A.M. License Renewal Application and Final Safety Evaluation Report (SER) for the Shearon Harris Nuclear Power Plant, Unit 1 (Open) (JS/PW)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff and Carolina Power & Light Company regarding the license renewal application for the Shearon Harris Nuclear Power Plant, Unit 1, and the associated NRC staff's final Safety Evaluation Report (SER).

   Members of the public may provide their views, as appropriate.

10:00 - 10:15 A.M. * BREAK *****

3) 10:15 -12:15 P.M. Status of Resolution of Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance" (Open) (SB/DAW/DB)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff and PWR Owners Group regarding the staff and industry activities associated with the resolution of GSI-191.

   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. Selected Chapters of the SER Associated with the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification Application (Open/Closed) (MLC/HJV)
   - 4.1) Remarks by the Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the

NRC staff and General Electric - Hitachi Nuclear Energy (GEH) regarding selected Chapters of the NRC staff's SER With Open Items associated with the ESBWR design certification application.

Members of the public may provide their views, as appropriate.

[NOTE: A portion of this session may be closed to protect information that is proprietary to GEH and its contractors pursuant to 5 U.S.C. 552b (c) (4).]

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

5) 3:30 - 4:00 P.M. Quality Assessment of Selected Research Projects (Open) (DAP/HPN)

5.1) Remarks by the Subcommittee Chairman

5.2) Discussion of the draft final report on the quality assessment of the NRC research projects on: FRAPCON / FRAPTRAN Code work at the Pacific Northwest National Laboratory (PNNL), and NUREG/CR - 6943, "A Study of Remote Visual Methods to Detect Cracking in Reactor Components."

6) 4:00 - 5:15 P.M. Historical Perspectives and Insights on Reactor Consequence Analyses (Open) (WJS/HPN)

Discussion of the draft White Paper prepared by Dr. Nourbakhsh, ACRS Senior Technical Advisor, regarding historical perspectives and insights on reactor consequence analyses.

5:15 - 5:30 P.M. ***** BREAK *****

7) 5:30 - 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

7.1) License Renewal Application for the Shearon Harris Nuclear Power Plant, Unit 1 (JS/PW)

7.2) Status of Resolution of Generic Safety Issue - 191 (SB/DAW/DB)

7.3) Selected Chapters of the SER Associated with the ESBWR Design Certification Application (MLC/HJV)

FRIDAY, OCTOBER 3, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

8) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (WJS/CS/SD)

9) 8:35 – 9:30 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed) (WJS/EMH/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.

[NOTE: A portion of this meeting may be closed pursuant to 5 U.S.C. 552b (c) (2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.]

10) 9:30 - 9:45 A.M. Reconciliation of ACRS Comments and Recommendations (Open) (WJS/CS/AFD)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

11) 9:45 – 10:00 A.M. Subcommittee Reports (Open)

11.1) Report by and discussions with the Chairman of the ACRS Subcommittee on Materials, Metallurgy, and Reactor Fuels regarding Proposed Supplemental Pressurized Thermal Shock Rule (10 CFR 50.61) that was discussed during the Subcommittee meeting on October 1, 2008. (WJS/CLB)

11.2) Report by and discussions with the Chairman of the ACRS Subcommittee on Reliability and PRA regarding the draft final NUREG-1855, “Guidance on the Treatment of Uncertainties in Risk-Informed Decisionmaking,” that was discussed during the Subcommittee meeting on September 30, 2008. (GEA/HJV)

10:00 - 10:15 A.M. * BREAK *****

12) 10:15 – 11:30 A.M. Preparation for Meeting with the Commission on November 7, 2008 (Open) (WJS, et al. /EMH, et al.)
Discussion of proposed topics for meeting with the Commission
on November 7, 2008.

11:30 - 12:30 P.M. * LUNCH *****

13) 12:30 - 7:00 P.M. Preparation of ACRS Reports (Open)
Continued discussion of proposed ACRS reports on:

13.1) License Renewal Application for the Shearon Harris Nuclear Power Plant, Unit 1 (JS/PW)

13.2) Status of Resolution of Generic Safety Issue - 191 (SB/DAW/DB)

13.3) Selected Chapters of the SER Associated with the ESBWR Design Certification Application (MLC/HJV)

SATURDAY, OCTOBER 4, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14)	8:30 - 1:00 P.M. **(10:30-10:45 A.M. BREAK)**	Preparation of ACRS Reports (Open) Continue discussion of the proposed ACRS reports listed under Item 13.
15)	1:00 - 1:30 P.M.	Miscellaneous (Open) (WJS/EMH) Discussion of matters related to the conduct of Committee activities and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTES:

- During the days of the meeting, phone number 301-415-7360 should be used in order to access anyone in the ACRS Office.

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

- Thirty five (35) hard copies and one (1) electronic copy of the presentation materials should be provided to the ACRS in advance of the briefing.

September 22, 2008

<u>COLOR CODE – 556TH ACRS MEETING</u>

License Renewal Application for the Shearon Harris Nuclear Power Plant, Unit 1	JS/PW	***Goldenrod***
Status of Resolution of Generic Safety Issue- 191	SB/DAW/DB	***Salmon***
Selected Chapters of the SER Associated with the ESBWR Design Certification Application	MLC/HJV	***Green***

(Letters may be added or deleted after Committee consideration.)

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September 22, 2008

OVERTIME SCHEDULE*
FOR THE 556th ACRS MEETING

Thursday, October 2, 2008
Theron Brown
Sherry Meador
Jessie Delgado

Friday, October 3, 2008
Theron Brown
Sherry Meador
Natalie Mitchell-Funderburk

Saturday, October 4, 2008
Theron Brown
Sherry Meador

* Any changes to above schedule should be checked with Ed Hackett.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
556th MEETING

LICENSE RENEWAL APPLICATION AND FINAL SER for the
SHEARON HARRIS NUCLEAR POWER PLANT, 1

Thursday, October 2, 2008

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……………………………………………………….
Cognizant ACRS Member: John Stetkar
Cognizant ACRS Staff Engineer: Peter Wen

Advisory Committee on Reactor Safeguards
Review of License Renewal of
Harris Nuclear Plant (HNP)
October 2, 2008
Rockville, MD

- PROPOSED SCHEDULE-

Cognizant Staff Engineer: Peter Wen Peter.Wen@NRC.GOV (301) 415-2832

Topics	Presenters	Time
Opening Remarks	J. Stetkar, ACRS	8:35 am - 8:40 am
Staff Introduction	B. Holian, NRR	8:40 am – 8:45 am
HNP Introduction	C. Burton, CP&L	8:45 am – 8:50 am
HNP License Renewal Application • Site Description • Operating History • Major Plant Improvements • Plant Performance • License Renewal Project • Open Item Resolution • Subcommittee Follow-up Items	Carolina Power & Light Company (CP&L)	8:50 am – 9:25 am
NRC Staff Review Summary NRC Onsite Inspection Results	M. Heath, NRR C. Julian, Region II	9:25 am – 9:50 am
Committee Discussion	J. Stetkar, ACRS	9:50 am – 10:00 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.

- Presenters to bring in 40 copies of the presentation slides prior to the meeting and save the presentation in the slide projection laptop.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
REVIEW OF LICENSE RENEWAL OF
HARRIS NUCLEAR PLANT
OCTOBER 2, 2008

- STATUS REPORT -

PURPOSE

The purpose of this meeting is to review the License Renewal Application (LRA) for Harris Nuclear Plant (HNP) and the associated final Safety Evaluation Report (SER) dated August 21, 2008 (Reference 1). The full Committee will hear presentations by and hold discussions with representatives of the U.S. Nuclear Regulatory Commission (NRC or the staff) and the applicant, Carolina Power & Light Company (CP&L). The License Renewal Subcommittee previously met to discuss this application and the draft SER with open items on May 7, 2008.

BACKGROUND

The HNP is a single unit facility located approximately 16 miles southwest of Raleigh, North Carolina. The HNP is a Westinghouse PWR with a licensed power of 2900 MWt. The NRC issued an operating license for HNP on January 12, 1987, which is in effect until midnight on October 24, 2026.

DISCUSSION

By letter dated November 14, 2006, CP&L submitted the LRA for HNP (Reference 2) in accordance with Title 10, Part 54, of the *Code of Federal Regulations* (10 CFR Part 54).

CP&L is requesting renewal of the operating licenses for HNP, for a period of 20 years beyond the current expiration date of October 24, 2026. The staff reviewed the LRA for HNP in accordance with the NRC regulations and NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LP), dated September 2005. Title 10, Section 54.29, of the *Code of Federal Regulations* (10 CFR 54.29) provides the standards for issuance of a renewed license.

In addition to NUREG-1800, the staff used the following guidance in its review:

(1) NUREG 1801, Rev. 1, "Generic Aging Lessons Learned (GALL) Report," September 2005 (Reference 3)

(2) Regulatory Guide 1.188, "Standard Format and Content for Application to Renew Nuclear Power Plant Operating Licenses," endorses NEI 95-10, Rev. 6, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54, The License Renewal Rule," September 2005

On May 7, 2008, the ACRS Subcommittee on Shearon Harris License Renewal heard presentations from the staff and the applicant. At that time, the staff had issued a draft SER with one open item. The draft SER presented the status of the staff's review of information submitted by the applicant through February 19, 2008. In addition to the one open item, it contained 2 confirmatory items, 3 proposed license conditions, and 35 commitments. Since then, the one open

item has been resolved and the 2 confirmatory items were closed with two additional commitments added. As a result, the final SER contains three license conditions and 37 commitments.

OPEN ITEM RESOLUTION

As a result of its review of the LRA, including additional information submitted through February 19, 2008, the staff had earlier identified one open item (OI). An item was considered open if, in the staff's judgment it did not meet all applicable regulatory requirements at the time of the issuance of the draft SER with Open Items. The staff assigned a unique identifying number to the OI. As discussed below, this OI has been closed by the staff.

OI-2.2: (SER Section 2.2 - Plant Level Scoping Results)

During the May 7, 2008, ACRS Subcommittee meeting for Shearon Harris License Renewal, there was only one open item in the staff's draft safety evaluation report (SER). This open item was related to the scoping classification of feedwater regulating valves and bypass valves. HNP included these valves in the LRA within the scope of license renewal per the criteria of 10 CFR 54.4(a)(2), because it believed that these valves are not safety-related components. However, the staff believed that these valves fulfill a safety-related function; therefore, they should be included in scope under 10 CFR 54.4(a)(1).

By letter dated May 30, 2008, the applicant provided a discussion of the proposed resolution to this open item and LRA Amendment 8. In the response, the applicant explained that based on its current licensing basis (CLB), regulatory guidance (Section 15.1.5 of SRP-LP), and NUREG-0138, these feedwater regulating valves and bypass valves are properly classified as nonsafety-related components and should be scoped within the license renewal under 10 CFR 54.4(a)(2). The CLB and the regulatory guidance indicate that applicant can take credit for these nonsafety-related valves as a backup to safety-related feedwater isolation valves in mitigating main steam line break. In addition, LRA Amendment 8 revises LRA Section 2.3.4.6 to add a description of the feedwater system safety function to isolate feedwater flow following certain main steam line break accidents. This amendment also adds the following 10 CFR 54.4(a)(1) function to the table of intended functions: supports isolation of feedwater flow following certain main steam line breaks, supports the containment isolation function, and supports post-accident monitoring. This inclusion resolves the staff's concern that the feedwater isolation function was not identified as an intended function of the feedwater system in the original LRA Section 2.3.4.6.

The staff has reviewed the applicant's response and the cited documents. The staff agreed with the applicant's assessment that the feedwater regulating and bypass valves are properly classified as nonsafety-related components and should be scoped for license renewal under 10 CFR 54.4(a)(2). The staff also verified that the amendment to the LRA does not bring additional components within the scope of license renewal. Therefore, the staff determines that Open Item 2.2 is closed.

SUBCOMMITTEE FOLLOW-UP ITEMS

In addition to the OI discussed above, there are three items that the applicant did not provide complete answers to the members' questions during the Subcommittee meeting. The applicant stated that it would provide complete answer during the full committee meeting. These follow-up items include:

1. Information regarding manhole inspections, water accumulation, corrective actions, historical operating experience, cables design and capabilities, and potential systems that could be affected with a cable failure (due to submerged cables).

2. Information regarding the containment spray valve chamber corrosion, groundwater intrusion, which were identified during past refueling outages and the root cause of the identified "degradation."

3. Information on the HNP cooling system diagram, visual images of the Main and Auxiliary Reservoirs and a description of the cooling system.

PROPOSED LICENSE CONDITIONS

Following the staff's review of the LRA, including subsequent information and clarifications from the applicant, the staff identified three proposed license conditions:

1. The first license condition requires the applicant to include the FSAR supplement required by 10 CFR 54.21(d) in the next FSAR update, as required by 10 CFR 50.71(e), following the issuance of the renewed license.

2. The second license condition requires future activities identified in the FSAR supplement to be completed prior to the period of extended operation (PEO).

3. The third license condition requires all capsules in the reactor vessel that are removed and tested meet the requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the staff prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the staff as required by 10 CFR Part 50, Appendix H.

COMMITMENTS

Commitments made by the licensee are listed in detail in Appendix A to the SER. The licensee made 37 commitments related to the aging management programs (AMPs) to manage aging effects of structures and components to be implemented before the PEO.

ONSITE AUDIT AND REGIONAL INSPECTION ACTIVITIES

In support of the staff's review of the LRA for HNP, an NRC project team conducted three onsite audits to review the AMPs, AMRs, and TLAAs and issued a report dated March 26, 2008 (Reference 4). Also, an inspection was performed by Region II inspectors. The inspectors reviewed the screening and scoping of non-safety related systems, structures, and components in AMPs, and observed activities that were related to license renewal. The Region II inspection report is dated September 10, 2007 (Reference 5).

EXPECTED COMMITTEE ACTION

The Committee will review this matter and may provide a report during the October 2-4, 2008 ACRS meeting.

References

1. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report, Related to the License Renewal of Shearon Harris Nuclear Power Plant, Unit 1,” dated August 21, 2008, (ML082340985).
2. Letter dated November 14, 2006, from Cornelius Gannon, CP&L to U.S. Nuclear Regulatory Commission, transmitting the Application to Renew the Operating License of Shearon Harris Nuclear Power Plant, Unit 1, (ML063350267).
3. U.S. Nuclear Regulatory Commission, NUREG-1801, Volumes 1 & 2, Rev 1, “Generic Aging Lessons Learned Report,” September 2005.
4. NRC Staff Audit Summary Report, dated March 26, 2008. (ML080800243)
5. NRC License Renewal Inspection Report 05000400/2007007, dated September 10, 2007, (ML072530894).

Progress Energy

MAY 3 0 2008

SERIAL: HNP-08-055
10 CFR 54

U. S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

Subject: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400 / LICENSE NO. NPF-63

LICENSE RENEWAL – RESOLUTION OF OPEN ITEM AND LICENSE RENEWAL APPLICATION AMENDMENT 8

References:
1. Letter from Cornelius J. Gannon to the U. S. Nuclear Regulatory Commission (Serial: HNP-06-136), "Application for Renewal of Operating License," dated November 14, 2006
2. Letter from P. T. Kuo (NRC) to Robert J. Duncan II, "Safety Evaluation Report with Open Items Related to the License Renewal of the Shearon Harris Nuclear Power Plant, Unit 1," dated March 18, 2008

Ladies and Gentlemen:

On November 14, 2006, Carolina Power & Light Company, doing business as Progress Energy Carolinas, Inc., requested the renewal of the operating license for the Shearon Harris Nuclear Power Plant, Unit No. 1, also known as the Harris Nuclear Plant (HNP), to extend the term of its operating license an additional 20 years beyond the current expiration date.

By letter dated March 18, 2008, the Nuclear Regulatory Commission (NRC) issued the Safety Evaluation Report with Open Items Related to the License Renewal of the Shearon Harris Nuclear Power Plant, Unit 1. Section 1.5 of the report identified an Open Item for which resolution had not been achieved at the time the report was issued. This letter submits: (1) information to close the Open Item and (2) changes to the HNP License Renewal Application (LRA) that support closure of the Open Item.

Therefore, this letter contains two enclosures. Enclosure 1 provides the information required to resolve the Open Item. Enclosure 2 is a table that provides the changes to the LRA; the changes constitute LRA Amendment 8. The information provided in this letter does not affect any of the License Renewal Commitments; therefore, any actions identified in this letter should be considered intended or planned actions; they are included for informational purposes but are not considered to be regulatory commitments.

Progress Energy Carolinas, Inc.
Harris Nuclear Plant
P. O. Box 165
New Hill, NC 27562

A126
NRR

Please refer any questions regarding this submittal to Mr. Roger Stewart, Supervisor - License Renewal, at (843) 857-5375.

I declare, under penalty of perjury, that the foregoing is true and correct (Executed on MAY 30 2008).

Sincerely,

Christopher L. Burton

Christopher L. Burton
Director – Site Operations
Harris Nuclear Plant

CLB/mhf

Enclosures:

1. Resolution of License Renewal Open Item
2. Amendment 8 Changes to the License Renewal Application

cc:

Mr. P. B. O'Bryan (NRC Senior Resident Inspector, HNP)
Ms. B. O. Hall (Section Chief, N.C. DENR)
Mr. M. L. Heath (NRC License Renewal Project Manager, HNP)
Ms. M. G. Vaaler (NRC Project Manager, HNP)
Mr. L. A. Reyes (NRC Regional Administrator, Region II)

Resolution of License Renewal Open Item

Background

On November 14, 2006, Carolina Power & Light Company, doing business as Progress Energy Carolinas, Inc., submitted a License Renewal Application (LRA) and requested renewal of the operating license for the Shearon Harris Nuclear Power Plant, Unit No. 1, also known as the Harris Nuclear Plant (HNP), to extend the term of its operating license an additional 20 years beyond the current expiration date.

By letter dated March 18, 2008, the Nuclear Regulatory Commission (NRC) issued the Safety Evaluation Report (SER) with Open Items Related to the License Renewal of the Shearon Harris Nuclear Power Plant, Unit 1. Section 1.5 of the report identified an Open Item (OI) for which resolution had not been achieved at the time the SER was issued. A statement of the OI is provided below followed by information required to resolve it.

Open Item 2.2 (Section 2.2 Plant Level Scoping Results)

In LRA Section 2.3.4.6, Feedwater System, the applicant did not identify the feedwater isolation function in scope for license renewal under 10 CFR 54.4 (a)(1). In Section 15.1.5 of the applicants FSAR, it states that the feedwater isolation valves and regulating valves provide a safety-related function, isolation of feedwater in the event of a main steam line break. The staff's position is that the FSAR description of the feedwater isolation and regulating valves meet the criteria defined by 10 CFR 54.4(a)(1). In response to RAI 2.1.1.2-1, the applicant stated that based on their evaluation the feedwater regulating and bypass valves, these valves do not meet the license renewal definition of safety-related as stated in 10 CFR 54.4(a)(1); however, the components are included within the scope of license renewal for 10 CFR 54.4(a)(2). The staff found the applicants answer to RAI response 2.1.1.2-1 inconsistent with 10 CFR 54.4 (a)(1).

In RAI 2.3.4.6-2 the staff asked the applicant to further evaluate the classification of this equipment and justify their position. The applicant's response, dated January 22, 2008, maintains that these valves are important to safety, but are not safety-related; therefore, they only meet the criteria of 10 CFR 54.4(a)(2). The staff's position remains that the main feedwater regulating and bypass valves, by definition, fulfill a safety-related function; therefore, they should be included in scope under 10 CFR 54.4(a)(1). In addition, the function to provide main feedwater isolation should be included in scope under 10 CFR 54.4(a)(1) for Section 2.3.4.6, to include the main feedwater isolation valves and the regulating and bypass valves. This is OI 2.2.

Response to Open Item 2.2

In the previous response to this issue, provided by Progress Energy letter to the NRC (Serial: HNP-08-006), dated January 22, 2008, the function and qualifications of the Feedwater Regulating Valves (FRVs) and FRV Bypass Valves within the HNP Current Licensing Basis (CLB) were recounted. The HNP CLB permits the backup isolation of main feedwater flow to steam generators, under certain assumed conditions following a postulated main steam line

rupture, to be accomplished by non-safety grade components. This aspect of the CLB is based on guidance provided by the HNP Nuclear Steam System Supplier (Westinghouse) and regulatory guidance documented in the Acceptance Criteria of Section 15.1.5 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." Application of this guidance to system design is more fully discussed in Issue No. 1 of NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff." NUREG-0800 was used by the NRC staff in the licensing review of HNP, and Section 10.4.7 of NUREG-1038, "Safety Evaluation Report Related to the Operation of Shearon Harris Nuclear Power Plant Units 1 and 2," November 1983, states that the FRVs are Quality Group D components, i.e., not fully safety grade.

Based on the functions and qualifications of the FRVs and FRV Bypass Valves in the HNP CLB, the License Renewal scoping evaluation for these components concluded that they were properly included in scope under 10 CFR 54.4(a)(2), that is, non-safety related systems, structures, and components (SSCs) whose failure could prevent the satisfactory accomplishment of any safety related functions.

The HNP License Renewal Application states that the License Renewal methodology used at HNP is consistent with the approach recommended in NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," Revision 6. When applying the guidance for scoping safety related SSCs of NEI 95-10, the HNP scoping methodology focused on: (1) assuring that the definition used during design and operation of HNP to identify safety related SSCs was consistent with the definition provided in 10 CFR Part 54 and (2) identifying cases where SSCs had been classified safety related but did not actually perform a safety related function in accordance with 10 CFR 54.4(a)(1). Guidance for these activities is provided in NEI 95-10; however, guidance is not provided for the situation where a safety related function could be accomplished by non-safety grade components as permitted by the HNP CLB. Therefore, the HNP scoping process relied on the CLB and not on the guidance provided in NEI 95-10 to make a final determination that these SSCs were not safety related under 10 CFR 54.4(a)(1). This is considered to be an exception to the scoping methodology described in Section 3.1.1 of NEI 95-10.

Based on the preceding discussion, the License Renewal conclusion that these non-safety grade components meet the criteria of 10 CFR 54.4(a)(2) is consistent with the HNP CLB.

Based on the above response, a License Renewal Application amendment is required to include the feedwater isolation function in the discussion of the Feedwater System and to note that the scoping methodology for HNP takes an exception to the guidance of NEI 95-10.

Amendment 8 Changes to the License Renewal Application

<table>
<tr><th>Source of Change</th><th>License Renewal Application (LRA) Amendment 8 Changes</th></tr>
<tr><td>Resolution of Open Item-2.2</td><td>
Add the following sentence after the first sentence of paragraph two of LRA Section 2.0:
<blockquote>Exceptions to the scoping guidance of NEI 95-10 has been taken consistent with the HNP Current Licensing Basis.</blockquote>
Revise LRA Subsection 2.3.4.6 on Page 2.3-248 by revising the first full paragraph to read:
<blockquote>The system serves no safety function, with the exceptions of containment isolation integrity and termination of feedwater flow following certain main steam line break accidents; and is therefore generally classified as non-safety related. The portion of the system classified as safety related is the portion from the feedwater header check valves to the Steam Generators.</blockquote>
Also, revise the first row of the table of intended functions in the fourth paragaph to read:
<table>
<tr><td>10 CFR54.4(a)(1) Functions</td><td>Supports isolation of feedwater flow following certain main steam line breaks,
Supports the containment isolation function, and
Supports post-accident monitoring.</td></tr>
</table>
</td></tr>
</table>

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON THERMAL HYDRAULIC PHENOMENA
GENERIC SAFETY ISSUE 191

SEPTEMBER 23, 2008

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Cognizant ACRS Member: Sanjoy Banerjee
Cognizant ACRS Staff: David Bessette

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
556th ACRS MEETING
OCTOBER 2, 2008
ROOM T-2B3

UPDATE ON GS-191, "Assessment of Debris Accumulation on PWR Sump Performance" - Status and Future Activities (OPEN)

3. 10:15 -12:15 P.M.

Status of Resolution of Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance" (Open) (SB/DAW/DB)

3.1 Remarks by the Subcommittee Chairman

Briefing by and discussions with representatives of the NRC staff and PWR Owners Group regarding the staff and industry activities associated with the resolution of GSI-191.

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

NOTES:

- 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

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE ON THERMAL HYDRAULIC PHENOMENA

SEPTEMBER 23, 2008

-STATUS REPORT-

PURPOSE

As part of continuing actions to resolve GSI 191, "Assessment of Debris Accumulation on PWR Sump Performance," the objective of this meeting is to review the progress made by the PWR Owners Group and the staff towards resolving GSI 191 and associated Generic Letter 2004-02. afety evaluation of the report. The outcome is expected to be a report from the Committee at the 556th meeting, October 2-4 10-12, 2008 .

BACKGROUND

In 1979, as a result of evolving concerns about the adequacy of PWR recirculation sump designs, the staff opened Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance." To support the resolution of USI A-43, RES undertook a research program whose results are documented NUREG-0897, "Containment Emergency Sump Performance." The resolution of USI A-43 was based on this work and was issued to industry in Generic Letter 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," December 3, 1985.

At the time, the regulatory analysis did not support imposing new sump performance requirements for PWRs or BWRs. However, Generic Letter 85-22 concluded that the 50-percent blockage assumption under which most plants had been licensed, as identified in Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," Revision 0, should be replaced with a more comprehensive requirement to assess debris effects on a plant-specific basis. The 50-percent screen blockage assumption did not require a plant-specific evaluation of the debris-blockage potential. This could result in a nonconservative estimate of screen blockage effects. The staff updated the NRC's regulatory guidance, including Section 6.2.2 of the Standard Review Plan (NUREG-0800) and Regulatory Guide 1.82 to reflect the USI A-43 findings described in NUREG-0897.

Following the resolution of USI A-43 in 1985, several events occurred in BWRs:

- On July 28, 1992, at Barsebäck-2, the spurious opening of a PORV led to the plugging of two containment spray system suction strainers with mineral wool. The operators to shut down the spray pumps and back-flushed the strainers.

- At Perry-1, on January 16, 1993, ECCS strainers were plugged with particulate matter from particulates in the suppression pool. Soon after, on April 14, an ECCS strainer was plugged with glass fiber from ventilation filters that had fallen into the suppression pool. On both occasions, the affected ECCS strainers were deformed by excessive differential pressure created by the debris plugging.

- On September 11, 1995, at Limerick-1, following a manual scram due to a stuck-open SRV, operators observed fluctuating flow and pump motor current on the A-loop of suppression pool cooling. These indications were attributed to formation of a thin mat of fiber and sludge which had accumulated on the suction strainer.

In response to these events, several generic communications were issued including:

- Bulletin 93-02, Supplement 1, “Debris Plugging of Emergency Core Cooling Suction Strainers,” February 18, 1994;

- Bulletin 95-02, “Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode,” October 17, 1995; and

- Bulletin 96-03, “Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors,” May 6, 1996.

These bulletins requested that BWR licensees implement procedures, maintenance, and plant modifications to minimize the potential for clogging of ECCS suction strainers during a LOCA. Implications concerning the adequacy of PWR sump designs were also raised. Contrary to USI A-43, the amount of debris generated by a high-energy line break (HELB) could be greater, finer, more easily transportable, and combinations of debris (e.g., fibrous material plus particulate material) could result in substantially greater clogging. This led to establishing GSI-191.

On June 9, 2003, following its assessment of GSI-191, the staff issued Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors.” The Bulletin requested PWR licensees assure compliance with long term cooling requirements using mechanistic based analysis, or to describe any interim compensatory measures to minimize risk until the analysis could be completed. The staff recognized that it might be necessary for licensees to undertake complex evaluations to determine whether regulatory compliance existed in light of the concerns identified in the bulletin, and that the methodology needed to perform such evaluations might not be available. As a result, information on regulatory compliance was not requested in the Bulletin.

Instead, the Bulletin indicated that the staff was preparing a generic letter that would request that information, namely, Generic Letter 2004-02. Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors,” was issued on September 13, 2004. This GL requested all PWR licensees:

1. Use an NRC-approved methodology to perform a mechanistic evaluation of the potential for post accident debris blockage and operation with debris-laden fluids to impede or prevent recirculation; and

2. Implement plant modifications or other corrective actions which the evaluation identifies as necessary to ensure system functionality.

On December 6, 2004, the staff issued a safety evaluation of a May 28, 2004, NEI report (ML041550279) on “Pressurized Water Reactor Containment Sump Evaluation Methodology.” The NEI report, in conjunction with the staff’s safety evaluation (ML043280007 and

ML043280008), satisfied requirement 1 above for a method acceptable to the staff for evaluating PWR sump performance as requested in the GL.

By September 2005, licensees responding to GL 2004-02 indicated that 66 of the 69 PWRs developed plans to replace sump screens; the other 3 PWRs had already done so. Several ACRS members and staff visited Salem recently to look at the newly sump screens, which are about a factor of 50 to 100 times larger than the old ones. The deadline for implementing the plant modifications was set at December 31, 2007. Many plants were subsequently granted extensions to early 2008.

DISCUSSION

Based on new information during the efforts to resolve GSI-191, the staff determined that previous guidance used to develop current licensing basis analyses did not adequately model sump screen debris blockage and related effects. The deficiencies in the previous guidance potentially result in analytical errors that could cause degrade ECCS performance. Therefore, the staff is revising its guidance for determining the susceptibility of PWR recirculation sump screens to debris blockage as well as downstream effects during design basis accidents requiring recirculation operation of the ECCS or containment spray.

The staff is using information received in response to GL 2004-02 to: (1) formulate an auditing approach for verifying that PWR licensees have resolved the concerns identified in the generic letter; (2) assist in determining which PWR licensees would be subject to the audits; (3) provide confidence that unaudited licensees have addressed the concerns identified in the generic letter; and (4) assess the need for and guide the development of any additional regulatory actions that may be necessary to address the adequacy of ECCS and containment spray recirculation.

A number of licensees are performing tests on the new screen to obtain information on blockage and bypass. A key number being used for bypass is 1 ft^3 per 1000 ft^2 of screen area. There are still uncertainties in chemical effects and the downstream deposition of bypass material either in the lower plenum, the core inlet or in spacer grids.

EXPECTED COMMITTEE ACTION

The Full Committee is expected to write a report on GSI 191 status during its October 2-4, 2008 meeting.

REFERENCES

1. "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," WCAP-16793-NP, Rev. 0, June 4, 2007, ML071830044

2. Letter to Gordon Bischoff, Manager, Owners Group Program Management Office Transmitting the Draft Safety Evaluation for Pressurized Water Reactor Owners Group Topical Report WCAP-16793-NP, Revision 0, “Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid” from Stacey L. Rosenberg, Chief, Special Projects Branch, Office of Nuclear Reactor Regulation, March 7, 2008, ML080600876

3. Letter to Anthony R. Pietrangelo, Nuclear Energy Institute, from William H. Ruland, Director, Division of Safety Systems, Office of Nuclear Reactor Regulation, "Draft Conditions and Limitations for use of Westinghouse Topical Report WCAP-16793-NP, Revision 0, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," February 4, 2008, ML080180415

4. Letter to Gordon Bischoff, Manager, Owners Group Program Management Office, Westinghouse Electric Company, from Tanya M. Mensah, Senior Project Manager, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation, "Request for Additional Information re: Pressurized Water Reactor Owners Group Topical Report WCAP-16793-NP, Revision 0, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid, September 10, 2007, ML072410036

5. Evaluation of Downstream Sump Debris Effects in Support of GSI-191, Timothy S. Andreychek, et al, WCAP-16406-P Draft Revision 1: Section 9.0 (cold leg injection, hot leg injection, fiber, particulates, etc.) May 2006, ML061580184

6. NUREG-0897, Revision 1, "Containment Emergency Sump Performance - Technical Findings Related to USI A-43," October 1985.

7. "Pressurized Water Reactor Sump Performance Evaluation Methodology, Revision 1 NEI 04-07, ML050550138.

ATTACHMENTS

ATTACHMENT 1
Minutes of the last thermal hydraulic subcommittee meeting on GSI 191

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE ON THERMAL HYDRAULIC PHENOMENA
MARCH 19, 2008
ROCKVILLE, MARYLAND

INTRODUCTION

The ACRS Thermal Hydraulic Phenomena Subcommittee met with representatives of the NRC Staff, Westinghouse, and the PWR Owners Group. The purpose was to review the PWR Owners Group report, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous, and Chemical Debris in the Recirculating Fluid," WCAP-16793-NP, and the staff's draft safety evaluation. During the meeting, the Subcommittee heard presentations by and held discussions with Westinghouse, the PWR Owners Group, its consultants, and the NRC Staff. David Bessette was the Designated Federal Official. The meeting was convened by the Chairman at 8:30 am and adjourned at 6:00 pm.

ATTENDEES

ACRS Members/Staff

Sanjoy Banerjee, Chairman
Said Abdel-Khalik, Member
Dennis Bley, Member
Michael Corradini, Member
Otto Maynard, Member

Tom Kress, Consultant
Graham Wallis, Consultant
David Bessette, Designated Federal Official

Staff

James Beall
Ellery Coffman
Ralph Landry
Robert Litman, Staff Consultant
Paul Klein
William Ruland
Mike Scott

Industry

Tim Andreychek, Westinghouse/PWROG
Matt Brands, AREVA, PWROG
William (Art) Byers, Westinghouse
Mo Dingler, Westinghouse/PWROG
David Fink, Westinghouse
Brett Kellerman, Westinghouse/PWROG
Kevin Menames, Westinghouse/PWROG
Paul Pyle, Westinghouse/PWROG
William Rinkacs, Westinghouse/PWROG
Gordon Wissinger, AREVA/PWROG

AGENDA

1.	Introductory Remarks by the Chairman	Sanjoy Banerjee
2.	Update on GSI-191 Status and Future Activities	Mike Scott, NRR
3.	PWR Owners Group Presentation WCAP-16793-NP	Mo Dingel, Westinghouse W Tim Andreychek, W

4. NRC Staff Presentation on Draft SE for WCAP-16793-NP Ralph Landry
Paul Klein

5. Subcommittee Discussion

1. CHAIRMAN'S INTRODUCTORY REMARKS

The Chairman opened the meeting, stating that its purpose was to review the PWR Owners Group report, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous, and Chemical Debris in the Recirculating Fluid," WCAP-16793-NP, and the staff's draft safety evaluation.

2. Status Summary - Mike Scott, NRC Staff

Mr. Scott presented an overview of status and activities underway to resolve GSI-191. Generic Letter 2004-02 is the staff's regulatory vehicle to resolve GSI 191. GL 2004-02 requested that licensees determine their plant specific debris generation and transport and make any changes to plant design necessary to ensure long term cooling in the recirculation mode. The plants were asked to respond by the end of 2007 but most requested and received extensions to that date. All actions will not be complete until 2009. In the meantime, in view of low probability and the installation of the new screens, the staff considers the schedule for GSI 191 resolution satisfactory.

The plants responded to the generic letter by installing new plant screens 20 to 100 times larger in area. Several plants have sent submittals indicating they are in compliance with long term cooling requirements. NRR believes the risk of sump screen blockage has been reduced considerably but that significant uncertainties still exist with respect to debris generation, transport, settling, and screen deposition.

Installation of the new screens has preceded testing in most cases which creates some risk, however, the Commission and industry believed it was important not to delay installation of the new screens. If further changes were found to be necessary it would be in the area of reducing fibrous insulation or changing the buffer.

Testing of the new screens has uncovered new information rather than being just proof testing. An important factor in the results is the way in which debris is introduced into the flow, such as order of addition. It has taken industry some time to perform the tests and to resolve staff comments. Prof. Wallis indicated some of the testing should, therefore, be termed research.

The staff was not ready to talk about details of test conduct at this meeting because information is still arriving and being evaluation. The staff has performed nine audits thus far on how licensees have conducted test programs and made changes to the plant design including removal of fibrous insulation as well as the changing screens. In general the licensees were following staff guidance. One area that was generally lacking was chemical and downstream effects. Licensees were waiting on WCAP-16530 and WCAP-16793. Several additional audits will be performed in 2008.

The staff is making some attempt to bin the plant based on sump screen vendor because of similarities in screen design but there are differences in design by the same vendor based on containment configurations.

Testing shows that particulates pass through the sump screen when they are introduced before fibers. When the order is reversed, the fibers tend to catch on the screen and in turn filter the particulates.

Prof. Wallis questioned the basis for the bypass of 1 ft^3 of debris per 1000 ft^2 of screen area. (The 1 ft^3 of debris refers to as-manufactured fiberglass density of ~38 kg/m^3 (2.4 lb/ft^3) as compared to the density of ordinary glass is ~2500 kg/m^3). Mr. Klein indicated that the number came from a wide range of testing of screens. The data are from four or five different sump screen vendors. The uncertainty in the number was said to be ±10%. Supporting data were not presented.

It was noted that information exists from testing in Germany and Japan, as well as from other facilities located in foreign countries. It was also noted that German plants do not add a buffer to the sump water. About ½ dozen US plants have changed the species of buffer used.

At the last subcommittee meeting in May 2007, industry discussed three approaches to dealing with uncertainties in addressing the problem: 1) removing fibrous insulation; 2) test results showing a smaller zone of influence; and 3) calculate settling.

BWRs are being re-evaluated by RES to see whether lessons learned from GSI-191 have any regulatory impact or not.

3. PWR Owners Group Presentation WCAP-16793 - Mo Dingler, Tim Andreychek, Westinghouse

Mr. Dingler began the presentation by addressing bypass testing sponsored by the PWR Owners Group. Tests were run with: particulate only; fiber only; and mixtures of the two. The findings were that only small amounts of very small length fibers passed through the screens. The distribution of fiber lengths used in the tests was from jet impingement experiments. Some of the data date from the 1970s. Prof. Wallis raised a question about whether aged insulation is more brittle than new material. Mr. Dingler noted that the insulation is encapsulated in the plant.

Mr. Andreychek's presentation addressed chemical and downstream effects based on the topical report WCAP-16793. Westinghouse is preparing a revision to this report based on requests for additional information from the staff and its responses.

PWR fuel assemblies have inlets designed to prevent foreign material from entering the core. The hole diameter of the inlet assemblies is ½ to 2/3 the screen mesh size. WCAP-16793 considered collection of debris at the core inlet, in spacer grid, and on cladding surfaces. A peak cladding temperatures of 800F was adopted as the success criteria since it corresponded to the experimental data base from autoclave experiments run for 30 days. The acceptance criterion for cladding surface deposits is 50 mils to avoid bridging 110 mil subchannels.

Early on following shutdown, approximately 250 gpm of makeup is required to replace losses from boiling. By 30 hours this number is reduced to ~150 gpm as decay heat decreases. Chairman Banerjee inquired about how much resistance would be necessary to restrict flow to the core or past spacer grids until the makeup rate was equal to the boiling rate. This calculation has not been performed.

Westinghouse did perform a calculation that showed that one unblocked fuel assembly, or its equivalent, out of ~200 is sufficient to keep the core covered. Prof. Wallis noted that predictions of possible resistance and tolerable resistance are not found in WCAP-16793. Dr Landry stated that tests were run with a large accumulation of debris using a prototypic core inlet plate that produced a pressure drop of less than 1/3 psi. It was noted that the sump screen area is about 100 times larger than the core inlet flow area.

Chairman Banerjee inquired about chemical gels that might accumulate at the core entrance and cause large flow resistance. Mr. Andreychek indicated that such gels, if they formed, would be captured by the sump screens. According to Mr. Klein, Westinghouse assumed that anything that got past the screens was transported to the core. The solubility of aluminum hydroxide was noted to be a function of temperature. Testing by at least one licensee supports the assumption that precipitates will be transported to the core.

Tests run at Argonne National Laboratory showed very high flow resistance developing at the core inlet. The tests were run a room temperature and, therefore, did not properly treat solubility of aluminum hydroxide early in an accident where sump temperatures are high. However, as long term cooling proceeds, sump temperatures will drop and precipitation may be expected to increase. Also between the sump and the core the flow passes through a heat exchanger.

Dr. Kress inquired whether the analyses indicated any difficulties with either boron dilution or concentration and precipitation. Dr. Landry replied that it was being addressed on a plant specific basis.

Mr. Andreychek indicated that the lengths of fibers that passed through the screen were determined to be in the range of less than 750 μm (0.03 in) to 2000 μm (0.08 in). Profs. Abdel-Khalik and Wallis indicated that the value of 1 ft^3 ±10% per 1000 ft^2 seemed surprising given the uncertainties in the inlet flow conditions and whether the tests were really prototypic. What may be conservative for screen blockage may be nonconservative for bypass. Experiments indicate that most of the integrated debris that gets through the screen is bypassed early and that bypass drops off as debris accumulates on the screen.

An experimental facility was shown that employed a single prototypic bundle inlet nozzle, ~1/3 height fuel bundle array, and a lower plenum representation. Tests run in this facility indicated settling in the lower plenum and some accumulation on the inlet nozzle and lowest spacer grid.

One-dimensional radial heat transfer calculations were performed for fuel rods to investigate effects of postulated debris deposits on cladding temperature. A 1979 test report prepared by NUKON was referenced. These tests including heating a rod up to 2200F and quenching it in a fibrous slurry. Other testing included two-hour tests with nucleate boiling and film boiling. The fibers did not adhere to the cladding.

In WCAP-16793 it is assumed that particulates or solutes that get through the screen are transported to the core. In the core, where ever boiling is occurring this material is assumed to deposit on the cladding according to their concentration in the water. A conduction study was performed to evaluate the effect on heat transfer of deposition of precipitates on the cladding. The pre-existing layer of ZrO_2 and crud were also included. Prof. Wallis indicated that the discussion in WCAP-16793 of this work was very unclear.

LOCADM is a spreadsheet code developed to calculate chemical deposition on cladding (method from WCAP-16530-NP-A). The conduction calculations were done with ANSYS. Surface heat transfer used coefficients from COBRA/TRAC. The conductivity of ZrO_2, (2.79 W/m-C), was determined from Halden data ["Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," WCAP-15063-P-A pp 53-56, ML003735390; and EPRI TR-107718-P1 and P2]. This data had a range of approximately ±1 W/m-C. The value used for conductivity of crud was 0.5 W/m-C. The value used conductivity of post-LOCA scale deposits was 0.2 W/m-C and was varied from 0.17 to 1.5 W/m-C, from literature survey of boiler scale deposits. The conductivity of precipitates was varied by about a factor of 10 from 1.6 to 0.17 W/m-C. A study was also done on reduction in flow at spacer grid but no results presented. A maximum clad temperature of less than 800F was adopted as the acceptance criterion.

Paint and coatings from containment were considered as sources of zinc, epoxies, etc. to the sump water.

Boric acid was said not to precipitate according to current accepted licensing calculations.

Westinghouse 2-loop upper plenum injection plants issues are much the same as for cold leg injection.

Profs. Wallis and Banerjee noted that many of the slides included qualitative statements rather than quantitative results. Mr. Scott noted that the staff's safety evaluation took note of many conservative assumptions in the Westinghouse evaluation in concluding the overall approach to be conservative.

Mr. Andreychek continued with a description of the COBRA/TRAC analyses. One assumed 99.4% blockage corresponding to 216/217 fuel assemblies. In the other all central fuel assemblies were blocked (K ramped up at the time of switch over to recirculation 20 min into the calculation from 1.5 to 1 x 10^9, no blockage of core periphery. If only one fuel assembly, out of 193 in the core, is open to flow, the inlet flow velocity is ~4 ft/s, about 1/3 the normal pumped, full power flow velocity. While there is a flow path between the core barrel and the core baffle, with holes in the baffle wall leading to the core, this flow path is not credited in the analyses.

Prof. Wallis and Chairman Banerjee indicated that Westinghouse did not provide evidence with which to evaluate the applicability of COBRA/TRAC for this type of analysis. Mr. Dingler acknowledged this was so and stated the staff performed independent calculations in lieu of this information.

The Chairman inquired why no calculation was done assuming a uniform mat with a loss coefficient of, perhaps 10^4 or 10^5.

Once water enters the core from below, COBRA/TRAC calculates good lateral flow across the core from gravity force.

Within the core region conduction heat transfer calculations were made for a subchannel within a spacer grid. The ΔT between the coolant and the cladding hot spot was calculated to be ~150F. The calculation was done assuming a 50 mil deposit of post-LOCA scale deposits, while Westinghouse indicated that conservative calculations show layers ~10 mil. Mr. Klein indicated that the LOCADM spread sheet that was used to calculate scale deposits on cladding based on local boiling rates. It predicted ~10 mils deposit on the worst case rod. Prof. Wallis

noted that there were a few cases of inverse solubility with fluid temperature such as calcium. In response to a question by the Chairman, Mr. Andreychek indicated that boiling in the core stops when injection is switched from the cold leg to the hot leg, ~4 to 9 hours into the accident.

There followed a discussion of what would happen for various configurations of debris collection at the core inlet, within the core, and at the core exit. Mr. Dingler and Mr. Andreychek indicated that experiments did not produce compressed mats at the core entrance that might lead to a substantial flow resistance. Mr. Scott added that the experiments indicated substantial settling in the lower plenum with most of remaining debris forming a rather loose permeable layer at the core entrance.

Mr. Steve Smith from the staff returned to the question of the experimental evidence for the amount of sump screen bypass. Based on four tests, a conservative estimate for the bypass was 1.3 ft^3 per 1000 ft^2 of screen area (other tests have been performed but the data have not been archived). The bypass rate decreased substantially with time as fibers built up on the screen. The approach velocity to the screens was ~0.1 inch/s. Mr. Scott indicated that licensees must show that debris settles upstream of the screens in order to take credit for it. Otherwise, debris is assumed to stay in suspension until reaching the screen. In the four tests discussed there was not substantial settling. The Chairman noted that bypass should be dependent on formation of a mat on the sump screen and that under prototypic conditions this may form more non-uniformly than in the experiments, which is expected to increase bypass.

The question of fiber size distribution was revisited. Mr. Dingler noted that the method used at Wolf Creek was to lay individual fibers under a microscope next to a ruler to measure their length.

4. NRC Staff Presentation on Draft SE - Ralph Landry, Paul Klein

Dr. Landry began by introducing Paul Klein of the staff, who was responsible for review of chemical effects, along with Robert Litman, a consultant. Dr Landry discussed the buildup of scale on cladding surfaces, as calculated using LOCADM. Licensees are likely to credit treatment of chemical and downstream effects described in WCAP-16793 as conservative, but they are expected to verify that this is indeed so. Prof. Wallis enquired how the staff would respond if licensees submitted much different estimates for the same process, such as bypass. Mr. Klein replied that the staff attempts to reconcile the reasons for the differences in that situation. There are one or two principal reviewers for the submittals so differences between licensees will most probably be caught and resolved.

Concerning the effects of blockage at the core inlet, the staff performed audit analyses using RELAP5 and TRACE, and also with FLUENT for core flow distribution. The TRACE calculation included inlet flow blockage of 95%. This calculation showed an increase of maximum clad temperature of 10F over no blockage. There was also little difference in void fraction between blocked and unblocked core entrance cases. The blockage assumption was similar to the COBRA/TRAC calculation in that the 95% blockage case, for example, treated 95% of the core entrance area as completely blocked.

Dr Landry revisited the four Continuum Dynamics Inc (CDI) tests run to study debris behavior in the lower plenum and core. The inlet nozzle was supplied by Calvert Cliffs. Atypicalities of the test included the short, 1½ foot length, unheated plastic bundle. The pressure drop across the core entrance was ~1/3 psi.

In response to a question by Prof. Wallis, the staff has not yet obtained reports or data on debris deposition experiments performed by AREVA in Germany.

Prof. Wallis noted that in tests performed by Argonne National Laboratory included chemical effects, which the CDI tests did not, a factor of 100 increase in flow resistance was observed. Mr. Klein replied that CDI ran some tests for a licensee that included 100% of the WCAP surrogate chemical and observed a pressure drop of 0.2 psi to 2.6 psi across the core inlet, based on a flow rate for a hot leg break, which is much higher than flow necessary to remove decay heat. Mr. Klein indicated that the WCAP-specified amounts of sodium-aluminum-silicate and aluminum-oxyhydroxide are very conservative.

Dr Landry indicated that the Westinghouse conduction calculations to determine peak cladding temperature treated the thicknesses of the oxide, crud, and scale layers all in a conservative manner to maximize their thicknesses. The prior clad oxide layer was assumed to be at the 17% limit; the crud layer was taken to be 100 μm, and a 1.27 mm (50 mil) scale layer, which is 5 times more that the maximum calculated from LOCADM.

The question of whether the lower plenum participates as a mixing volume with the core with respect to boron was raised. The idea is that if the core entrance is so severely blocked as to inhibit gravity driven upflow of water into the core, then it might inhibit downflow as well. Prof Wallis noted the volume occupied by material transported to the vessel is very small. Dr Landry indicated that some plants do not take credit for mixing between the core and lower plenum. In response to questions by subcommittee members, Mr. Andreychek stated that the question of whether such mixing will take place should hinge mainly on the relative density of the water in the core, whose density is increasing due to boron concentration, and the higher density of the colder water (~90F) entering the lower plenum compared to the ~260F water in the boiling region of the core.

Mr. Klein reported on the chemical effects review. He noted that the staff last briefed on WCAP-16530 in May 2007. The safety evaluation of this report is now complete. A peer review of this report was performed and Mr. Klein indicated some of the questions raised became RAIs. Based on review of WCAP-16530 the staff decided to perform confirmatory tests at Argonne National Laboratory on head loss and at Southwest Research Laboratory on chemical effects. The staff concluded that the WCAP was conservative in assuming all dissolved aluminum and silicon combines with phosphates to rapidly precipitate. Passivation of aluminum was not considered. The WCAP surrogate was based on thermodynamic equilibrium. The Argonne tests indicated that using the WCAP surrogate could produce quite high flow resistances. The Argonne loop has a vertical orientation so settling was precluded and all the precipitates ended up at the strainer surface.

Tests performed by some of the screen vendors for licensees have concluded that the WCAP-specified quantities of precipitates formed are conservative. Even making the effort to keep precipitates in suspension, settling occurs. Also, the deposition with a horizontal arrangement is nonuniform compared to the vertical Argonne section.

WCAP-16793 uses the assumptions from WCAP-16530 for downstream chemical effects such as cladding deposition. In response to a question by Prof. Wallis, Dr Litman indicated that since boiling occurs preferentially in pits and crevices, and cladding surface conditions with crud are porous, scale tends to collect preferentially in these surface imperfections. Once scale is

formed on cladding surfaces, the WCAP assumes it stays put and does not re-dissolve. The scale is taken to be a sodium-aluminum-silicate with a conductivity of 0.2 W/m-C (~0.1 BTU/hr-ft-F), which falls in the range of insulating materials.

There followed a discussion between the subcommittee and the presenters regarding conclusions. Prof. Wallis inquired about how the LOCADM evaluation model had been validated and demonstrated to be conservative. The last slide in the presentation gave a comparison of the model with an experiment taken from a paper. The Chairman requested that the staff provide the reference.

An example of a LOCADM calculation was included in WCAP-16793-NP and reviewed by the staff. It was for a large 3188 MWt plant with high fiber (7000 ft^3) and a large quantity of calcium silicate insulation (80 ft^3). The maximum scale deposit calculated over 30 days was 10 mils. The maximum clad surface temperature after the start of recirculation was 324F, well below the 800F acceptance criteria.

Dr Landry indicated that Westinghouse will provide the members of the PWR Owners Group with a guidance document on implementing WCAP-16793-NP as well as LOCADM model. Licensees must demonstrate applicability of existing sump screen data or perform their own tests. Core inlet flow blockage must be shown to not affect prior boron mixing analyses.

Prof. Wallis inquired how a licensee could decide what would be acceptable to the staff based on what appears in WCAP-16793-NP. Mr. Scott indicated that a bounding pressure drop from the CDI experiments, based on scaled transport of ~22 ft^3 of fiber and 1389 lb of particulates was equivalent to 10 inches (~1/3 psi) of water (tests did not include chemical effects). Together with the COBRA/TRAC calculation of 99.4% inlet flow blockage, this was an example of providing assurance.

Prof Wallis and the Chairman inquired about providing guidance to the licensees concerning chemical effects which are considered to deposit on the screens but not at the core entrance. Mr. Dingler indicated that tests have shown much less deposition at elevated temperatures than at room temperature. Temperature at the core inlet will become significantly lower than at the screen because the water passes though the RHR heat exchanger. The Chairman noted that the screen and the core inlet function as two screens in series, the one with ~50 times the surface area of the other, so a coupled evaluation model should be carried out.

Drs. Litman and Landry asserted that stored energy in the vessel is not gone for a day, however, no calculations have been done to support the assertion. The Chairman stated that the core inlet temperature has to be known to support the assertion that compounds remain in solution and do not deposit at the inlet.

The Chairman and Prof. Wallis noted that WCAP-16793 does not provide methods to follow, but sets forth conservative assumptions that when followed should bound the outcome, and if these assumptions are followed, then the outcome meets acceptance criteria. Prof. Wallis gave the example that there is no description of where fibers will end up. There is, however, the calculation that the entrance to the core can be almost completely blocked (99.4%) with no detrimental effect on peak cladding temperature.

The Chairman, referring to slide 43 from the PWR Owners Group presentation, requested an analysis of how much flow resistance at the core entrance would be necessary to reduce inlet flow below boiloff flow until the acceptance criteria of 800F would no longer be met.

Mr. Scott indicated that, as written on page 8 of the staff's safety evaluation report, the staff does not believe that a uniform bed will form at the core inlet.

Mr. Scott indicated the staff would address the question of whether bed formation at the core inlet affects whether or not the lower plenum functions as a mixing volume for boron.

Mr. Scott indicated that the question of in-vessel fluid temperatures would be investigated. He also indicated that the staff would look into acceptance criteria and methods of analysis for in-vessel chemical effects.

Mr. Dingler indicated that Westinghouse would send design geometry of the lower nozzle and spacer grids to Mr. Scott.

Mr. Scott indicated that there was not much additional testing beyond what was discussed during the meeting but that such testing may be needed. If confirmatory testing were sponsored by NRC, new results would take two or more years. Some information may be sought from licensees.

5. Subcommittee Discussion

Dr. Kress noted that downstream effects depend on bypass and there is no real technical basis for predicting how much bypass will occur. Dr. Kress added that validation of cross flow modeling in COBRA/TRAC should be provided. He concluded that substantial progress has been made in issue resolution with changing insulation and increasing the screen size. Dr. Kress expressed doubt that the debris distribution at the core inlet could ever be determined but the critical flow resistance to limit the inflow rate to the boiling rate should be analyzed.

Mr. Maynard stated that we will never have a definitive set of experiments and some qualitative decision making will have to be accepted. Any test data referenced in WCAP-16973 should be defended in the report as being applicable to the issue for which it is being used. The removal of debris-generating insulation is a move in the right direction.

Prof. Abdel-Khalik expressed reservations of the 1 ft^3 of debris per 1000 ft^2 screen area. The staff gave a number of 1.3 ft^3 which is outside the 10% uncertainty that was stated earlier during the meeting. The actual uncertainty could be an order of magnitude. An evaluation of the scaling and prototypicality of experiments is lacking, including whether important variables have been parametrically studied over the range of conditions expected in the plant. Prof. Abdel-Khalik agreed that the debris distribution at the core inlet is most likely indeterminate, but the flow resistance critical to limiting the inflow to the boiling rate should be analyzed.

The Chairman indicated that any data being generated by licensees to justify the 1 ft^3 of debris per 1000 ft^2 screen area should be obtained since is a very important number for downstream effects. The fact that one could block 99.4% of the inlet flow area and not affect core cooling was encouraging. The critical resistance of a distributed bed should be determined.

ATTACHMENT 2
Report by ACRS Consultant Tom Kress on the March 19, 2008 thermal hydraulic subcommittee meeting on GSI 191

ACRS received on April 3, 2008

March 25, 2008

Consultant's Report on the March 19, 2008 Thermal Hydraulic Subcommittee Meeting On GSI-191: Downstream Effects
T. S. Kress

A brief synopsis of the meeting's major points is as follows:

- Both sump screen clogging and downstream effects are plant specific issues.
  - The amount and type of debris generated via a LBLOCA
  - The description of the debris size and transport characteristics
  - Any chemical effects
  - The transport path to the sump screen and the associated thermal hydraulics
  - The size and type of sump screen installed and its capacity
  - The arrival time for the various debris types
  - The downstream geometry and injection locations/characteristics
  - The barriers to the core inlet and passage through the core
  - Etc.

- The individual plants have taken appropriate pragmatic actions:

  - increasing the screen sizes as much a possible in the given locations
  - minimizing the potential amount of debris
  - providing transport path barriers
  - eliminating sources of chemical effects (if possible).

- Based on limited data to date, a "rule-of-thumb" has been established that there will be about 1 ft3 of debris by-passing the screen (and, thus, becoming downstream debris) for every 1000 ft2 of screen surface area.

- Simulated core and debris testing has indicated that the core will remain within acceptance limits (e.g. 800 F and , .05 average cladding deposition and oxidation) even for a core inlet blockage of 99.5% of the inlet flow area. The acceptance criteria are based on ductility measurements after quench and thermal analyses for the fuel pins that include cross flow to distribute the flow throughout the core. Based on the above, the staff has agreed that unacceptable downstream effects are unlikely. I have the following comments on this issue.

1. The amount of by-pass debris depends strongly on the rule-of-thumb of 1 ft3 per 1000 ft2 of screen surface. This value needs improved technical justification for the different types of screens and geometries. The uncertainty in the value needs to be determined.

2. The result that the core meets appropriate acceptance criteria even with 99.5%

blockage is not necessarily a definitive argument that downstream effects are acceptable. New analyses are needed to assess the acceptable limiting debris friction coefficient (K) for which the core no longer meets the acceptance criteria. It will then be necessary to correlate K with the amount of mixed debris deposited uniformly of the screen surface in order to make a judgment as to the likelihood of this amount of debris being available as by-pass.

3. The ACRS needs to see the technical support for the cross flow determination.

4. Based on the margins in the current assessments, it is my opinion that there are no unacceptable downstream effects. On the other hand…..

5. I have been concerned from the outset of GSI-191 that the determination of the debris deposited on the sump screen is way too uncertain to rely solely on mechanistic calculations. I believe an appropriate uncertainty analysis is needed. I think this might indicate an unacceptable probability of loss of NPSH. Maybe it is too late at this juncture, but I would have liked to see a defense-in-depth requirement that the screens must be equipped with manually operated back-flow capability.

6. The staff has determined that there will be no deleterious effects on the mixing of boric acid and its function to maintain shutdown and prevent recriticality. At this time, I would like to raise an additional GSI-191 issue related to boric acid. As the water and boric acid solution boil off at low pressure, the boric acid tends to go preferentially with the steam to end up in undetermined locations in containment. This loss of boric acid along with the flow of fresh water into the core seems to me to provide the potential for dilution of the boric acid concentration and thus provide the potential for recriticality if the build up of xenon from iodine decay is not sufficient to offset this insertion of reactivity. The assessment of such a potentiality is relatively easy. The ACRS should recommend that the staff undertake such an assessment.

ATTACHMENT 3
CONSULTANT'S REPORT ON WCAP-16793-NP

Graham Wallis   March 25, 2008

I have studied the report. I also attended a meeting at NRC on March 19, 2008, at which the PWROG and the staff made oral presentations.

Since then I have looked over the RAIs and responses, which were not provided to us until March 20.   I also looked briefly at the staff's SER.

I do not have time to present a review of all of the matters that were presented. I will concentrate on the behavior of fibers, particles, and chemical precipitates.

GENERAL

The report and the responses to RAIs are full of qualitative assertions with little by way of substantial justification.

Experience throughout the study of GSI-191 has shown that there are many surprises. Guesses, mechanistic imaginations and even established correlations can be far from the mark when compared with comprehensive data taken with a questioning attitude.

The document provides little by way of guidance to a utility on how to predict the behavior and influence of fibers, particles and chemical precipitates.   The approach is to assert that they have small effects.

I have not had time to study the details of the SER. However, the staff's conclusion that "the application of the procedures and methods described in WCAP-16793-NP will provide an acceptable plant-specific evaluation of the plant's ability to remove long-term decay heat" appears optimistic in the light of the vagueness of much of the report and the RAI responses. It seems to be entirely based on the prediction by the code that if there is a pool of water supplied by almost any means (the report uses 99.4% blockage, with one assembly unblocked) then the core can be cooled.   This is a very useful conclusion, but I doubt if it can be used as an excuse for providing no specific models or recommendations for predicting the amount and properties of the fibers, particulates and chemicals, what they actually do, where they go, and what effects they have.

THE REPORT

Section 2 provides discussions of the effects of fibers and particulates. In the rest of the report there is little by way of substantiation of the assertions that are made there.

On p.2-1 it is stated that the formation of a "thin bed" 1/8 inch thick is sufficient to preclude flow to the core.   Later a figure of 1ft^3 of fiber bypass per 1000ft^2 of screen is mentioned. If we assume that this rule of thumb is valid, then with a screen area of 5000ft^2 the resultant 5ft^3 of fiber could make a layer over 1inch thick over a core flow area of 50ft^2.   It only takes a screen area of several hundred square feet to supply enough fiber to make the 1/8 inch thick layer.   So it might be concluded that a thin bed could block the core, an event which it is later asserted cannot happen.

p. 2-2   The amount of fiber bypass is said to be typically of the order of $1ft^3$ per $1000ft^2$ of sump screen area or less.   I have seen a couple of data points giving this sort of result, but the evidence for a general conclusion seems to be sparse. Surely the bypass depends on many things, such as the amount and character of the debris, the screen design, the flow in the vicinity of the screen and so on. It appears very unlikely that a simple rule of thumb will describe all cases.   Many tests performed by industry have been designed to maximize the head loss by creating a fiber mat over the entire screen. This has the effect of creating a barrier to bypass of small fibers and is probably not the condition that maximizes the bypass of fiber and particulates.

On the same page it is argued that the maximum debris size (fiber length?) passing through a screen hole of 0.10 inches is 0.11 inches. Ones shorter than 0.05inches are claimed not to bridge openings of the order of 0.05 to 0.08 inches at the core inlet and create a fiber bed.   It would seem that a few long ones could start the bed and others would join. Even the shorter ones could build up on corners and eventually bridge the gaps.

Fibers can cling to sharp corners, as on the grid spacers, and build up a layer. The experiments cited in the RAIs showed that a layer could form. Is there any substantial evidence to support the assertion that fibers that bypass the screen will not form a bed near the bottom of the core?

"debris that does collect will have some packing factor that will allow "weeping" flow through particulate debris buildup and into the core".   This is a qualitative statement, contradicted by the remark on the previous page about a thin bed.   What is needed is the development and validation of a robust technical approach to predicting what will happen in a specific plant. These general assertions are no help.

p.2-3   "This configuration provides less force to compact the collected fiber and particulate debris into a contiguous debris bed".   I doubt if this is true. Once a bed forms, the pressure drop through it becomes the major force tending to compact it and is independent of orientation.

"even if fibrous and particulate debris collected on the bottom of the fuel nozzle, the debris bed will not preclude water flow into the core".   This statement does not help a utility to compute the pressure drop through the particular bed that may form at the bottom of its core, given its particular strainer and all the various conditions determining fiber transport.

p.2-4 "It is clear from these discussions that adequate flow to remove decay heat will continue to reach the core even with debris from the sump reaching the RCS and core".

This conclusion cannot be based on the previous qualitative discussions about fibers and particles. It must rely on the computer calculations with flow entering only through one assembly. This result is impressive, but it does not cover the case where a bed forms over the entire base of the core.

The discussion of debris collection on fuel grids is a set of assertions unsupported by evidence.

p.2-5   The mention of "weeping" flow, debris "not becoming impenetrable" and "even a small amount of fluid flow" are too discursive to be convincing when dealing with such an important topic.   They also provide no useful guidance to the utility about what to calculate and how.

p.2-6 “water will pass through a pure fiber bed”. Yes, to some degree. But how much, under what conditions, and with how much head loss? How about particles and chemical precipitates depositing in the bed and raising the pressure drop by orders of magnitude, as has been found in some exploratory tests? How can predictions be made from such a general statement?

Precipitate is not only a concern because of possible plating on the fuel rods, but also for its enhancing effect on head loss through a fiber bed.

p.2-12 the discussion of “mixing volume” in the context of boric acid precipitation does not include the effect of a debris bed inhibiting mixing between the core and the lower plenum. Is this mixing credited? How is it influenced by debris? The statement “Mild core inlet blockage that would be expected for limiting boric acid precipitation scenarios would not preclude flow between the lower plenum and the core region” is very general and does not include any evaluation of what sort of flow distribution and mixing might occur in the presence of a fiber bed over part or all of the base of the core.

p.2-13 “total blockage in the upper plenum is not expected to occur”. What is the basis for this statement?

The conclusions on pp 2-15 to 2-17 would seem to require more substantial evidence.

RAI RESPONSES

I will run through the pages, mentioning some questions and answers. Several of these questions were asked at the March 19 meeting.

On page 1 of the first set of RAIs it is asked about the “basis of stating blockage of the core will not occur?” The answer is as vague as in the report itself.

p.5 A test of debris capture by a fuel assembly model is described. The fibrous debris collected at the first grid. “The test results clearly provide conservative amounts of fibrous debris”. There is no correlation of the pressure drop but it appears quite low. However, experience has shown that the way in which the debris arrives, addition of tiny amounts of chemical precipitates, and other experimental variables can have very large effects on pressure drop and bed behavior, sometimes increasing pressure drop by orders of magnitude. I doubt if this single test, which we have not reviewed, is comprehensive enough to provide assurance that these effects can be ignored. The tests do not appear to result in a method for making predictions and therefore are of little use to the utility which may have some other form of debris arriving in a different way.

p.8/9 It is stated that the NUREG/CR-6224 correlation does not apply. Then what is a utility supposed to use to make predictions?

p.10 It is asked if the blockage by fibers changes the boron mixing. The answer to this RAI is as vague as the one offered orally on March 19.

p.14 A question is raised about localized precipitation of boric acid and a locally blocked region. The answer is very qualitative.

p.16 “Please provide the flow loss coefficient that would occur”. The answer provides no coefficient, but responds to a different question.

p.32 “What is the maximum loading of fibers which could be concentrated within the core without adversely affecting core heat transfer?” The answer simply asserts “the effect would be small”.

p.34 “Please provide the guidance document that the utilities may utilize to perform specific assessments, for example:

1) maximum debris that enters the core and lower plenum

2) debris accumulation in the reactor vessel

etc;

The answer provides no guidance on these matters beyond asserting “fibrous debris will not block the core” and promising that a guidance document is being developed.

In the *second batch of RAIs* the first question is “Does the model that is being used account for surface irregularities capturing smaller debris than the span diameter?”

The answer is qualitative. Experiments with the real geometry and real debris are needed to validate any model and conclusions.

p.2 In reply to a question about particle agglomeration, the answer cites qualitative evidence from WCAP-16530-NP bench testing which showed that “freshly formed agglomerates were soft and easily torn apart”. Mechanistic and quantitative conclusions would be better.

ATTACHMENT 3: Committee Letters on GSI 191

July 19, 2004

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington DC 20555-0001

SUBJECT: PROPOSED DRAFT FINAL GENERIC LETTER ON POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PWRs

Dear Mr. Reyes:

During the 514th meeting of the Advisory Committee on Reactor Safeguards on July 7-9, 2004, we reviewed the staff's proposed draft final generic letter (GL) on the potential impact of debris blockage on emergency recirculation during design basis accidents at pressurized water reactors (PWRs). Our Subcommittee on Thermal-Hydraulic Phenomena reviewed this matter during a meeting held on June 22-23, 2004. During these reviews, we had the benefit of discussions with the NRC staff and its contractors, industry representatives, and members of the public. We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. A generic letter should be issued for implementation.

2. The staff should continue confirmatory research in areas where the technical basis of the guidance is uncertain, and on issues such as chemical and downstream effects that are not directly addressed by the guidance proposed by the Nuclear Energy Institute (NEI).

DISCUSSION

In our letter of February 20, 2003, we recommended issuance of the draft GL for public comment in order to initiate the process of gathering plant-specific information and requiring licensees to develop plans for resolving issues associated with potential sump screen blockage following a loss-of-coolant accident (LOCA).

We believe that a final GL should be issued to provide a consistent regulatory basis for action by the nuclear industry to ensure effective long-term reactor core cooling, in light of recent developments in the mechanistic understanding of the important phenomena.
We have not seen a final version of the GL. We understand that the staff is working on the details of the appropriate regulatory process, without changing the intent to resolve the technical issues expeditiously and practically.

The responses of licensees are likely to follow the guidance prepared by NEI and currently under review by the staff. We debated whether it was appropriate to issue a GL prior to completing the final NEI guidance document and the associated Safety Evaluation Report.

We have concluded that issuing a GL now will enable licensees to start the process of gathering information, planning activities, and performing preliminary analysis in anticipation of more complete analysis when guidance is available. It is important that the guidance be available in a timely fashion. We understand that it will be presented to our Thermal-Hydraulic Phenomena Subcommittee in August and to the full Committee in September 2004. At that time, the issue of chemical effects on screen blockage, which is not addressed in the NEI guidance, will not be resolved. We expect that the description of the risk-informed options in this guidance will be complete, including a clear description of what is meant by “mitigation capability” for breaks above a risk-justified transition break size. We look forward to discussing these issues with NEI and the staff.

The staff and the industry are also pursuing procedural approaches to make use of the robustness and adaptability of the installed systems to help ensure long-term cooling in the event of diminished recirculation pump capability. Such approaches, as suggested in our report of September 30, 2003, are an important aspect of risk-informing the sump blockage issue.

They may significantly reduce the risk associated with sump screen blockage, an effect which should be expressed in terms of quantitative measures so that its effectiveness can be assessed. Some technical issues raised by us previously have not yet been fully resolved. Research, such as the currently ongoing investigation of chemical effects and studies of downstream effects, is needed for the final resolution of the sump blockage issue. The staff should also initiate any research necessary to confirm that the guidance used by the licensees is adequate.

For example, there are still several different models for the “zone of influence.” We have questioned the technical basis for these models. Coupling an existing computational fluid dynamic code, such as FLUENT, to the steam-water properties available as a subprogram, would make it possible to model the homogeneous supersonic jets issuing from various break geometries. This work would show the shock wave patterns and mechanisms for energy dissipation and would be very helpful in evaluating the simplified zone of influence models.

Sincerely,

/RA/
Mario V. Bonaca
Chairman

References:

1. Proposed revised draft Generic Letter provided to ACRS on July 7, 2004 (Predecisional)

.

2. NRC Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors”, June 9, 2003.

3. Proposed Generic Communication: “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors”, 69 Fed. Reg 16,980, March 31, 2004.

4. Proposed revised draft Generic Letter provided to ACRS on June 15, 2004 (Predecisional).

5. Public Comment received from the Westinghouse Owners Group, May 27, 2004.

6. Public Comment received from Winston & Strawn (NUBARG), June 1, 2004.

7. Public Comment received from NEI, June 1, 2004.

8. Public Comment received from TVA, May 28, 2004.

9. Public Comment received from FPL, June 1, 2004.

10. Public Comment received from Duke Power, June 1, 2004.

11. Public Comment received from Westinghouse, May 27, 2004.

12. Public Comment received from UCS, May 20, 2004.

13. Public Comment received from Progress Energy, June 1, 2004
.

14. Public Comment received from Lanson R. Rogers, March 31, 2004
.

15. Public Comment received from Dominion, June 1, 2004.

16. Public Comment received from STARS, June 2, 2004.

October 18, 2004

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: SAFETY EVALUATION OF THE INDUSTRY GUIDELINES RELATED TO PRESSURIZED WATER REACTOR SUMP PERFORMANCE

Dear Chairman Diaz:

During the 516th meeting of the Advisory Committee on Reactor Safeguards, October 7-9, 2004 we met with representatives of the NRC staff, its contractors, and the industry to review the staff's draft safety evaluation (SE) related to Nuclear Energy Institute (NEI) Guidance Report (proposed document no. NEI 04-07), "Pressurized Water Reactor Sump Performance Evaluation Methodology" (Ref. 1 and 2). The guidance report and the associated SE are intended to describe a methodology that is acceptable to the staff for use by licensees in responding to Generic Letter (GL) 2004-02 (Ref. 3). Our Subcommittee on Thermal-Hydraulic Phenomena reviewed this matter during a meeting on September 22-23, 2004. We also had the benefit of the documents referenced.

Recommendations

1. The SE should not be issued in its present form. Both it and the NEI guidance contain too many technical faults and limitations to provide the basis for a defensible and robust long-term solution to Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on Pressurized Water Reactor (PWR) Sump Performance."

2. The faults and limitations in the present technical knowledge base need to be addressed so that acceptable guidance can be developed. The staff should develop sufficient understanding to determine either the uncertainty or the degree of margin resulting from the application of the methodology.

3. If licensees are to be responsible for filling gaps in the analytical and experimental data base, the staff should clearly state the agency's expectations for the necessary quality and acceptability requirements.

4. The risk-informed approach should be extended to treat the entire sequence of phenomena that lead from the break to the end effects on the pump net positive suction pressure (NPSH) and thus the effectiveness of recirculation cooling. This would provide a technical basis for application of the Regulatory Guide (RG) 1.174 process. It will require a quantitative assessment of model uncertainties related to the physical phenomena.

Discussion

GSI-191 is concerned with long-term cooling of the reactor core following a loss-of-coolant accident (LOCA). In the later stages of the accident scenario, water is drawn from the sump and recirculated through the core. The pumps that recirculate this water are protected by screens. There is concern that debris generated by the accident might accumulate on the screens sufficiently to compromise the NPSH of the pumps. The staff has taken several steps in its efforts to resolve GSI-191 In Bulletin 2003-01 (Ref. 4), which was issued on June 9, 2003, the staff requested that licensees "confirm their compliance with 10 CFR 50.46(b)(5) and other existing applicable regulatory requirements."

Regulatory Guide 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," was issued in November 2003. As we pointed out in our letter of September 30, 2003 the RG presented many requirements for analyzing the phenomena affecting sump performance, but provided little guidance on how to do the analysis.

In GL 2004-02, dated September 13, 2004, the staff requested licensees "to perform an evaluation of the ECCS and CSS recirculation functions in light of the information provided in this letter and, if appropriate, take additional actions to ensure system function."

The response of licensees to this GL depends on their ability to assess the constituent phenomena and their effect on the pump NPSH. Guidance on how to make these assessments is the subject of the NEI submittal and the staff's SE.

The staff has made an effort, both in its research programs and in its review process, to develop sufficient knowledge and judgment to evaluate the adequacy of these assessments. The staff believes that the NEI methodology, as modified by the staff in the draft SE, provides a conservative basis for evaluating PWR sump blockage. It is our judgment that too many gaps remain for a technically defensible resolution at this time.

The variety of technical challenges is large and there are unknowns and gaps in the knowledge base. For example, some basic methods include equations that contain incorrect physical descriptions of the phenomena. There are also questions about the extent of the supporting test data and the data's interpretation, and the guidance on implementing the proposed methods is vague and contains inconsistencies.

Risk-Informed Approach

In our report of September 30, 2003, we recommended that the staff investigate a risk informed approach to sump screen blockage. Such an approach is presented in Section 6.0 of the NEI guidance report and is essentially endorsed in Section 6.0 of the SE.

Although we welcome the use of risk in this context, it is not clear that the NEI approach, as modified by the SE, will have a significant practical impact. Plants will still have to compute the consequences of a large-break LOCA with respect to debris generation, transport, and deposition. It appears that there will only be a significant difference in what licensees may be required to implement if the requirement for "mitigation" is somehow related to risk, as well as to the assumptions that surround the traditional 10 CFR 50.46 LOCA requirements. Thus, we are recommending that the risk-informed approach be extended.

Besides the complex technical issues that make resolution difficult, the present deterministic requirements of 10 CFR 50.46 make it difficult for licensees to demonstrate compliance. The process of risk analysis was specifically developed to handle complex situations where uncertainties exist. We are recommending that risk information be developed and that the model uncertainties be quantified in the representation of the phenomena described in RG 1.82, Rev. 3. This would provide a technical basis for application of the RG 1.174 process.

Technical Errors

We were surprised to find significant technical errors of a fundamental nature in the analytical knowledge base supporting the guidance.

For example, the zone-of-influence (ZOI) model is based on the ANSI/ANS 1988 standard. There are several inconsistencies and errors in the models described in the standard, as discussed in References 7 and 9. The "impingement pressure" is undefined. The assumed flow pattern does not correspond to observed and computed patterns for supersonic jets. The conditions in a free jet and in a jet impinging on a large target appear to be mixed up. The analysis of the area of an "asymptotic plane" is based on an unrealistic representation of the physics and inappropriate one-dimensional approximations which can be used to calculate a variety of results, spanning a factor of four. In addition, the density at this fictional "asymptotic plane" is evaluated as if the fluid were at rest, whereas in reality it is flowing at a high Mach number.

We also identified significant basic errors in the NUREG/CR-6224 correlation and its use for evaluating head loss. These are described in References 5, 6, and 8.

Incomplete or Confusing Guidance

A "thin bed" is invoked at several places in the NEI guidance report and in the SE. There is no clear definition of what it is, how to predict its occurrence or its effects for different combinations of particulates and fibers, or how to "substantiate no formation" of it.

This is an example of an area where we consider the guidance to be confusing and inadequate. Appendix VIII of the SE addresses the thin bed effect in the context of calcium silicate (CalSil). The definition given is:

"The thin-bed effect refers to the debris bed condition in a fibrous/particulate bed of debris whereby a relatively high head loss can occur due to a relatively thin layer of debris, by itself or embedded as a stratified layer within other debris, because the bed porosity is dominated by the particulate and the bed porosity approaches that of the corresponding particulate sludge."

This definition is qualitative, but it clarifies some uncertainties in the SE. If this thin layer can occur anywhere in the bed, it might, for example, form on top of a thick (say four inch) layer of fiberglass. In contrast, the guidance described in Appendix V of the SE appears to apply only to uniform beds of debris.

The final phrase in the definition is conjecture since there appears to be no definitive evidence of the cause of the phenomenon. In Los Alamos National Laboratory (LANL) test 6H, which is discussed in Appendix V, there was an anomalous head loss that increased by an order of magnitude over the course of two hours while the flow rate was kept constant. Though the correlation could be made to fit this single data point by adjusting the specific surface area to the unusually high value of 800,000 ft2/ft3, this process provides no basis for extrapolation to other conditions.

Although this appendix provides some discussion of what might be the cause of the thin bed effect, it is speculative. Page 9 of Appendix VIII appears to discuss experiments performed in the recent past and not formally reported. It is clearly a work-in-progress and is not sufficiently mature to form the basis of guidance that could have a large impact on all plants containing CalSil insulation. It is also unclear if the effect could occur with certain latent debris or with the debris from destroyed coatings, as well as with CalSil.

Another example of confusing guidance occurs on page vi of the executive summary in the SE, where the staff imposes the following exception:

"The [NEI guidance report] does not provide guidance for those plants that can substantiate no thin bed effect, which may impact head loss results and limiting break conditions."

It appears that plants with any CalSil insulation could have the thin bed effect and therefore the methodology will predict clogging of their screens. Plants that do not have CalSil cannot use the guidance.

There is some discussion of the thin bed effect on page 69 of the main text of the SE. It is stated in the SE that the head loss model is valid for thicknesses larger than 0.125 inch.

"Below this value, the bed does not have the required structure to bridge the strainer holes and filter the sludge particles."

This apparently indicates that one does not have to worry about beds with a thickness less than 0.125 inch thick. In other words, 0.125 inch is a criterion for a thick bed, as it defines a minimum thickness. On this basis, thin beds should be acceptable. At the bottom of page 69, it is stated that calcium silicate can form a bed without supporting fibers, which contradicts the above statement.

On page 70 we find that “sufficient conservatism should be used in estimating the quantities of fibrous debris available to form a thin bed.” Does this mean that one should assume that no fibers are required? Since all plants have fibers of some sort, even in latent debris, do they all exhibit the thin bed effect?

It is clear that the qualitative discussions in Appendix VIII of the SE need to be replaced by consistent and unequivocal guidance on how to deal with the possibility that a layer, or layers, in the deposit on a screen may result in a particularly high head loss. This needs to be supported by definitive experiments. Loose use of the term thin bed in the present version of the SE merely confuses an already uncertain situation.

Another example of inappropriate guidance is the staff’s agreement with NEI that it is conservative to assume uniform debris accumulation on all types and orientations of screens. This appears to be a step of faith. It does not consider the possibility of a thin bed over part of the screen, or the layering of debris to form a thin bed as one of the strata in the accumulated layers. Anomalous and unexplained phenomena, which may increase the head loss by an order of magnitude, have been observed in tests with CalSil, and may be due to nonuniform effects of this sort.

If a stratified layer can form within a thick bed, such a layer could form somewhere in the strata on the screen and cause high enough head loss to challenge the NPSH in any lant with enough calcium silicate insulation to provide a layer comparable to the layer in LANL's test 6H, namely 0.018 inch. This amounts to about a gallon of CalSil on a 100-square-foot screen. Since the fine particles of calcium silicate are readily transportable to the screen, this would essentially mean that no plant could tolerate use of any amount of this insulation.

Examples of Gaps in the Empirical Knowledge Base

Coatings

The effect on coatings of a two-phase jet issuing from the break is not well understood. While NEI suggests a damage pressure of 1000 psi for qualified coatings, the staff adopts the default conservative requirement that the zone of influence (ZOI) for coatings be a sphere with radius equal to ten times the break diameter. The staff also requires hat all unqualified coatings within the containment be assumed to be destroyed and entrained. These requirements appear to be arbitrary assumptions. They are said to be “conservative” but no rationale has been offered to support this claim. It appears that the requirements lead to the prediction of large amounts of particulate matter being generated and transported to the screens.

The nature and effects of coating debris are unknown. There is no guidance on how to compute head loss for coating debris, whether or not a thin bed effect will occur, or hether coatings are truly particulate, or actually flakes. If they are particulate, then supposedly a correlation such as NUREG/CR-6224 (if it were to be corrected) could be used. However, if the coatings are flakes, then a new model would have to be developed to account for their potential to behave like leaves on a street drain, overlapping and obstructing the flow paths in a way that is not described by the usual "specific surface area" models

Debris transport

There are many uncertainties in the modeling of debris transport. For example, one of the staff's assumptions, allowing only 15% of the debris to be held up in inactive pools, is based on model predictions for one specific plant and leads to the conclusion that much of the debris will reach the pool. This conclusion may not apply to other plants. A method to perform plant-specific calculations is sketched in very general terms but its implementation would be difficult.

Head Loss Correlations

While the NUREG/CR-6224 correlation, if it were to be modified to remove the technical errors, might be approved for use, the database does not contain enough data to etermine the parameters to use in the correlation under realistic conditions. Licensees are required to develop their own results for latent debris and coatings. For other materials, licensees are required to "ensure that [the correlation] is applicable." Since the database for several materials, such as CalSil, does not cover all plant conditions, licensees may have to carry out new experiments to obtain definitive data. We are recommending that the staff clearly state the agency's expectations for the necessary quality and acceptability requirements for these experiments.

Appendix V of the SE provides additional guidance on the use of the NUREG/CR-6224 correlation. The formulae provided for computing specific surface area are based on the assumption that the debris bed is uniformly mixed, which may not be the case following a LOCA. There are several mechanisms that may create dense layers that contribute to higher head loss than would be computed using uniform mixing, as discussed in Appendix VIII. Table V-5 provides validation ranges of sources for fibrous insulation debris.

No assessment is made of applicability to the LOCA context or of the very sparse data from some of the sources. For instance, an applicant who chooses to use the highest value of specific surface area for CalSil deduced in LA-UR-04-1227 (which can lead to an order of magnitude increase in head loss) will be basing all its predictions on a single, isolated, data point obtained at a single flow rate and temperature, with a single thickness, composition, and mode of formation of the bed, none of which may be representative of plant conditions.

Chemical and Downstream Effects

No definite guidance is provided for evaluating either chemical or downstream effects, both of which have may become of major importance as more knowledge is acquired.

The Committee will continue to work with the staff to develop solutions to the sump screen blockage issue.

Sincerely,

/RA/

Mario V. Bonaca
Chairman

Additional Comments from ACRS Members Graham B. Wallis and F. Peter Ford

We agree with the recommendations of our colleagues. The SER and NEI guidance documents contain too many technical faults and limitations to provide the basis for a long-term defensible and robust evaluation of the PWR sump performance. However, in the short term, there may be some practical actions that can be explored. For instance, the staff should encourage licensees to pursue, at an early stage, corrective actions that will be as independent as possible of known model uncertainties. These actions may include, for example; removing material that is known to manifest anomalous or particularly detrimental sump blockage results in tests (and whose removal does not introduce secondary detrimental effects); use of "double-jacketing"; demonstrating that materials such as coatings are proven by testing to be sufficiently robust that conservatism in assessing their vulnerability can be reduced; testing alternative filtering devices, such as debris catchers and active screens to the point where their performance can be realistically characterized empirically, or can be conservatively bounded.

Such actions in this initial phase could be followed by a more complete evaluation of the technical analytical problem (as outlined in this ACRS letter) and the implementation of long-term risk-informed solutions.

Additional comments by ACRS Member Graham B. Wallis.

I agree that practical steps should be explored and long-term solutions developed. However, this responds only to part of the purpose of GL 2004-02, which also requests that licensees perform an evaluation in the short term. In the absence of definitive guidance for making this evaluation, the staff needs to develop a success path that will avoid wasteful iterations while guidance is developed. One approach might be to break up the process into a set of phases, each of which is based on the best available

methods, results that are clearly evident despite uncertainties, and decisions that can be implemented within a realistic schedule.

To justify actions which may have a major impact on operating plants, the staff needs to do a better job of explaining the rationale for regulatory decisions, particularly when the technical bases and assumptions are questionable.

References

1. U.S. Nuclear Regulatory Commission, Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, Nuclear Energy Institute Guidance Report (Proposed Document Number NEI 04-07), “Pressurized Water Reactor Sump Performance Evaluation Methodology”, September 2004

2. Nuclear Energy Institute, Guidance Report (Proposed Document Number NEI 04-07), “Pressurized Water Reactor Sump Performance Evaluation Methodology”, May 2004

3. U.S. Nuclear Regulatory Commission Generic Letter 2004-02: “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors”, September 13, 2004

4. U.S. Nuclear Regulatory Commission Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors”, June 9, 2003

5. Graham B. Wallis, “The NUREG/CR-6224 Head Loss Correlation”, September 3, 2004

6. Graham B. Wallis, “Flow Through a Compressible Porous Mat: Analysis of the Data Presented in Series 6 Tests Reported by LANL in LA-UR-1227", September, 2004

7. Graham B, Wallis, “The ANSI/ANS Standard 58.2-1988: Two-Phase Jet Model”, August 31, 2004

8. Sanjoy Bannerjee, “Review of Head Loss Prediction Across Sump Screens,” October 5, 2004

9. Victor Ransom, “Comments on GSI-191 Models for Debris Generation”, September 14, 2004

December 10, 2004

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: SAFETY EVALUATION OF THE INDUSTRY GUIDELINES RELATED TO PRESSURIZED WATER REACTOR SUMP PERFORMANCE

Dear Mr. Reyes:

Thank you for your letter of November 26, 2004, which responded to our letter of October 18, 2004, on the staff safety evaluation (SE) of the Industry Guidelines Related to Pressurized Water Reactor Sump Performance.

We appreciate the staff's desire to move ahead to resolve Generic Safety Issue-191, "Assessment of Debris Accumulation on Pressurized Water Reactor (PWR) Sump Performance." As licensees attempt to use the guidance, we anticipate that they will have to cope with several technical problems due to errors in the suggested methods. We disagree with your statement that the knowledge limitations are clearly identified and addressed in the SE. In our letter, we identified a number of these limitations. The purpose of this letter is to restate several of the limitations, and to respond to some of the staff's replies.

The head loss correlation in NUREG/CR-6224 (Ref. 1) is not entirely empirical, as claimed by the staff, but rests in part on the theoretical representation of two physical phenomena: the mechanical compression of the bed and the limit of this compression. The theoretical models for these phenomena are erroneous. Although some results may be predicted with apparent adequacy, the faulty models lead to some conclusions that are obviously at odds with reality.

For example, correlating bed compression with the pressure gradient is inconsistent with standard methods in the literature and cannot explain the compression of a fiber bed by the imposed pressure from a superposed particulate bed, as in the "thin bed effect." In addition, the NUREG/CR-6224 equation for the compression limit would predict that a fiber bed could be compressed up to the limiting particulate bed density even when there are no particles present, which makes no sense. The foundation of the correlation of data must be theoretically sound if the user of the guidance is to extrapolate a very limited range of data to real plant conditions.

The Committee commented in its letter that the effect on coatings of a two-phase jet is not well understood. The staff agreed "that the nature and effects of a two-phase LOCA jet on coatings are not well understood and that there is a lack of data on coatings." However, the staff still believes that the guidance is acceptable because of "precedents set by past applications approved by the staff and accepted by the ACRS or based on the staff approach of applying conservative assumptions to bound the unknowns." Unfortunately, because the phenomena are not well known, the uncertainties are also not well known, so the staff's "conservative assumptions" are only engineering judgment, without any technical basis.

We are pleased that the staff has alerted the American Nuclear Society to our technical comments on the 1988 ANSI/ANS standard (Ref. 2). However, the claim that Appendix I of the SE contains a "detailed evaluation" of this model is incorrect. Appendix I explains how to use the model, but repeats the technical errors contained in the model, such as the assumption of an "asymptotic plane" beyond which there are no supersonic effects, and the use of a stagnation density to describe a high-velocity stream. As a result, we have not seen convincing arguments that it is conservative to use the ANSI/ANS standard to determine the size of the zone of influence.

The staff claims that it is appropriate to assume that the debris bed is homogeneous, with the particles uniformly distributed through it. The staff also claims to supply guidance about the "thin bed effect," which is the extreme case where all the particles concentrate in a single layer.

These two arrangements of the debris are limiting situations of the general case in which various degrees of inhomogeneity occur; they cannot be true simultaneously. The guidance should address a wider range of possible inhomogeneities. It should allow the user to predict how much inhomogeneity occurs and the resulting head loss. There also needs to be better guidance on how the head loss evolves with time (as observed in experiments documented by NRC contractors), apparently because of the development of inhomogeneities, and on how extreme inhomogeneity can give rise to anomalously high head loss.

The guidance is also inadequate for evaluating downstream effects. It merely lists issues to be considered. It does not explain how to determine whether the issues are resolved, or how to perform an "integrated evaluation". Licensees will have to derive the acceptance criteria themselves.

There is also no useful guidance on chemical effects. The staff has only told the industry not to get caught by unexpected results from the ongoing experimental program. We continue to believe that both the SE and the Nuclear Energy Institute guidance document contain technical faults and limitations that will have to be corrected at some stage in order for the methods to be sufficiently robust and durable to support sound regulatory decisions.

Sincerely

/RA/

Mario V. Bonaca
Chairman

References:

1. NUREG/CR-6224, "Parametric Study of the Potential for BWR WCCS Strainer Blockage Due to LOCA Generated Debris," G. Zigler et.al., October 1995.

2. ANSI/ANS-58.2-1988, "Design Basis for Protection of Light-Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture," American Nuclear Society, October 6, 1988.

April 10, 2006

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: GENERIC SAFETY ISSUE 191 - ASSESSMENT OF DEBRIS ACCUMULATION ON PWR SUMP PERFORMANCE

Dear Chairman Diaz:

During the 530th meeting of the Advisory Committee on Reactor Safeguards, March 9-11, 2006, we considered several reports by the NRC staff regarding their efforts to resolve Generic Safety Issue 191(GSI-191), "Assessment of Debris Accumulation on PWR Sump Performance." The staff discussed licensee responses to Generic Letter 2004-02 (GL 2004-02), "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," and presented the results of efforts by the Office of Nuclear Regulatory Research (RES) to understand several phenomenological issues that have arisen as part of the GSI-191 effort, including chemical effects, downstream effects, and head loss correlations through debris beds. The results were presented to our Thermal-Hydraulics Phenomena Subcommittee on February 14-16, 2006. We had the benefit of presentations by and discussion with representatives of the NRC staff and members of the public. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. In response to GL 2004-02, many licensees plan to increase the size of their sump screens as quickly as feasible. Based on the current state of knowledge, we concur with this intent. However, it is not evident that this measure will be sufficient to resolve all long-term core cooling issues.

2. Results of prototypical experiments planned by industry to validate screen effectiveness will be difficult to extrapolate to plant conditions. Further work is required to provide the technical basis by which the staff can assess the adequacy of the planned modifications to the plants. Guidance should be developed to support the staff's review.

3. Recent research has revealed significant influences of particle/fiber mixtures and chemical reaction products on screen pressure drop for which improved predictive methods and guidance should be developed.

4. Increasing screen size to reduce the pressure drop may increase the amount of fine debris and chemical products that passes through the screen. Methods for predicting the quantity and properties of this bypassed debris should be developed. Potential adverse effects on downstream components, including pumps, valves, the core entrance regions, and the core itself, should be evaluated.

5. There has been some success at using adjustable parameters in an equilibrium chemistry model to match the chemical species that form in sumps. The methods should be validated further and guidance should be developed for their use.

6. The results of tests of coating debris formation and transport should be included in the assessment of core coolability as they become available. Future work should include the development of adequate predictive capability for the effects of coating debris on screen pressure drop and bypass.

OVERVIEW

At our meeting with the Commission on December 8, 2005, several Commissioners expressed the view that the sump screen issue should receive high priority. This was formally stated in the Commission's staff requirements memorandum of December 20, 2005: "... The ACRS shall make among its highest priorities its role in the resolution of GSI-191. ..." At the Commission meeting we indicated that we were waiting to hear status reports from the staff. W e have now received several reports, some of them preliminary, and this has enabled us to form an opinion on progress towards resolving GSI-191.

We have written previous letters on the sump screen issue. In particular we raised the matter of chemical effects and questioned some aspects of the NEI guidance which the staff had endorsed. The staff issued GL 2004-02 on September 13, 2004, and has received responses from all licensees. Though all licensees responded to the generic letter, the staff has concluded that none of the responses was complete. Gaps were evident in all important areas, particularly chemical and downstream effects. The staff has issued requests for additional information (RAIs) relating to several significant effects. Many licensees are finalizing plans to replace the screens before these RAIs are resolved.

While progress has been made in all areas of research, much remains to be done. These programs have produced significant results and are making important contributions to understanding the issues related to PWR sump performance. Many relevant physical and chemical phenomena are being explored. Assessments of other important effects may need to be added to the program.

This research has yet to lead to an ability to develop and validate predictive methods. Much of the work is exploratory in nature, in response to indications that existing analytical capabilities were incomplete and inadequate. The results from some programs are not yet available or are awaiting staff review.

The GL 2004-02 responses and recent research have raised new questions. Present plans by licensees to make hardware changes in their plants are driven by the need to reduce the potential for excessive head loss across sump screens during recirculation. Increasing the screen size will reduce this head loss, but the staff's ability to assess the adequacy of the reduction may be limited by uncertainties in the available knowledge base. In addition, downstream effects may be exacerbated by some screen designs and configurations. The staff needs effective means to evaluate these downstream effects and their influence on core coolability.

DISCUSSION

Industry Response to Generic Letter 2004-02

In general, licensees intend to address the sump screen issue by making a significant increase in the flow areas of the screens. Some designs may also have smaller openings and/or active debris removal mechanisms. Physical changes have already been made in some plants.

Modifications to almost all plants are planned to be completed by the end of calendar year 2007. Some licensees have requested extensions until the spring outage of 2008. Each of the five vendors of the new sump screens plans to undertake integrated-effect “proof tests” with screens or segments of screens to demonstrate the ability of the screens to accommodate the anticipated loading of debris with an acceptable pressure drop.

The prediction of debris formation, transport, and impact on core coolability is a very complex technical problem. A number of phenomenological issues must be addressed, either by the development of a predictive capability or by the implementation of engineering solutions that circumvent the more difficult issues. The industry is focusing on engineering approaches that maximize screen area to the extent practical, control of materials that affect the quantity and character of debris generation, and the control of sump chemistry to minimize chemical effects.

Regulatory Approach

The staff intends to undertake eight to ten audits of plant modifications. The scope of the audits will be expanded if the staff encounters problems with the technical adequacy of the planned resolutions.

Because of the “proof test” nature of the planned industrial testing program, it is essential that the staff have a level of understanding and a modeling capability for the underlying phenomena adequate to support their technical review of the licensee results. It is doubtful that the current understanding of these phenomena will be adequate to support such a review. The results of recent research have served to call into question some previous guidelines and assumptions without replacing them with validated, improved methods.

Research Efforts

Research is being performed to address the following phenomena:

- Chemical effects – experiments (Los Alamos National Laboratory (LANL) and Argonne National Laboratory (ANL)) and model development for speciation (Center for Nuclear Waste Research Activities (CNWRA))

- Head loss from debris buildup on screens – experiments (Pacific Northwest National Laboratory (PNNL)) and model development (RES)

- Downstream effects – experiments (LANL)

- Coating debris formation and transport – experiments (Electric Power Research Institute (EPRI), Naval Surface Warfare Center (NSWC))

We have seen only the preliminary results from some of these research efforts. It is premature for us to perform a comprehensive evaluation until all the work is complete. However, several research projects have developed important new quantitative information which reveals the significance of certain phenomena. Understanding of those phenomena has not yet been

established to the point where validated predictive tools are available. RES has set a target of the spring of 2006 to bring these activities to a conclusion. This schedule is unrealistic in view of the many unresolved issues.

Chemical effects

Exploratory integrated chemical effects tests (ICET) revealed that some species, particularly aluminum oxyhydroxide and calcium phosphate, can be produced under certain conditions. It was concluded that plant-specific evaluations would be required.

ANL is investigating the interaction between calcium silicate insulation (CalSil) and trisodiumphosphate (TSP), which forms calcium phosphate. A qualitative understanding of the chemical processes has been achieved. Studies of head loss on screens using debris quantities that duplicated earlier LANL tests with no chemical additives showed some variability.

When calcium phosphate was produced by adding TSP to CalSil, or calcium chloride to TSP, the pressure drop increased substantially. For example, in one test (ICET3-9) the pressure drop through a fiberglass bed was 0.14 psi at a flow velocity of 0.1 ft/s. When calcium chloride was added in stages to the solution of TSP, the pressure drop eventually rose to 5.2 psi at a flow velocity below 0.02 ft/s. Since the flow regime was probably laminar, for which pressure loss is proportional to flow velocity, this corresponds to an increase in bed resistance by a factor of about 200, amounting essentially to blockage of the screen. Similar results were obtained in Tests 1 and 2.

The results of chemical speciation prediction by codes using chemical equilibrium models and measured corrosion rates are encouraging over the range of species that have been studied. CNWRA found that some ICET results could be matched by adjusting the speciation parameters.

Head Loss Tests

PNNL has been conducting head loss tests with mixtures of fiberglass and CalSil in amounts corresponding to those used in earlier LANL tests. The results in some cases differ significantly from the results obtained by LANL. No distinct pattern is evident though some trends might be inferred. In an extreme case, when the constituents were introduced in a particular way, the head loss was roughly 100 times more than the head loss with a well-mixed debris bed of the same overall composition. These results indicate that the structure of the debris bed and the way in which it is formed can have a huge influence on the head loss.

Unless the assumption of a homogeneous bed can be justified, it will be necessary to develop an adequate model for these effects (for plants that intend to retain CalSil) or to find a way to scale them in the proof tests now planned by industry. The alternative of developing theoretical models for the way in which the bed builds up in different parts of the screen over time during a variety of accidents is probably unrealistic and may be beyond the capabilities of present state-of-the-art. RES has begun development of a theoretical model to predict the head loss in a nonhomogeneous debris bed. Substantiation and validation of such a model would be a major undertaking.

Downstream Effects

Tests conducted by LANL revealed that fine debris, of a size characteristic of the debris expected during energetic loss-of-coolant accidents (LOCAs), would pass through a typical sump screen under some conditions. Unless a debris bed has been established, most particles of CalSil and fine fiberglass pass through the screen. Significant quantities of reflective metallic insulation were observed to pass through under some conditions. In the absence of a detailed model for the history of debris bed development on a screen and the arrival of various constituents as functions of location and time, there are considerable uncertainties about how to apply such results to an actual plant. An order of magnitude calculation, with 5000 ft^3 of debris produced, indicates that about 6% of the debris would fill the typical lower plenum of a reactor vessel, if it settled there and was not transported to the core or filtered by debris catchers below the fuel. The larger the screen, the more open area there is likely to be through which fine debris can pass. Chemical reaction products are also likely to pass through open areas of the screen.

In reply to our subcommittee's questions about the effects of such debris on core coolability, the staff and representatives of the Westinghouse Owners Group (WOG) stated that they thought the core would be adequately cooled in a number of scenarios. However, they presented no physical models or analytical predictions to show a validated, quantitative basis for such conclusions.

Tests by LANL of debris transported to throttle valves have revealed a significant effect on pressure drop. Adequate predictive methods are therefore needed for the amount of this debris which actually reaches these valves, and for the resulting consequences.

Coatings

EPRI is conducting experiments on the formation of debris from qualified and unqualified coatings. The results were not presented at our meetings.

NSWC is conducting some basic tests of terminal velocity and transport of paint chips of various shapes, sizes, and composition. Guidance for use of these data remains to be developed.

What Is Missing

We are not aware of research efforts in several important areas.

The most significant omission appears to be an adequate understanding of the effects of the various debris species which enter the reactor vessel and reach the core. These effects are likely to depend on the LOCA scenario, particularly the location and size of the break, and on the screen design. Although guidance developed by the WOG describes several of the phenomena to be modeled to represent these effects, the WOG apparently leaves the evaluation to engineering judgment and ad hoc model development. Unless these effects can somehow be avoided, there is a need for a comprehensive set of validated tools for representing them. Developing the tools would involve significant experimental and model development efforts.

The proof tests being developed by industry to evaluate new screen designs involve the phenomena described earlier in this letter, as well as others. Synthesizing these evaluations into a defensible method for scaling test results to the actual LOCA scenario is no trivial matter.

We have yet to see scaling laws, methods of extrapolation, or theoretical representations (e.g. computational codes) which can make a convincing case that the test results can be applied to the actual plant. For example, one issue is how to use tests on a single module to predict the performance of an array of modules. The Office of Nuclear Reactor Regulation (NRR) may need to draw on further research results in order to evaluate submissions based on these proof tests.

Formation and transport of coating debris are being studied. We have not seen results of work on the effects of this debris on screen head loss. In view of the difficulty of predicting head loss with the existing mix of ingredients, and the surprises that have been encountered, it is necessary to establish a knowledge base for the effects of coatings on head loss by means of an adequate set of experiments and predictive methods.

Research has already revealed that the structure of a debris bed influences head loss and the bypass of fine material. As screens become larger and perhaps have more complex geometry, the variability of bed structure over the surface of the screen is likely to increase. Some areas, such as the base of vertical screens or the outer layers of multiple screens, may be covered by a pile of coarse debris, other areas may support "thin beds" that are blocked by chemical products or fine debris, while some areas may be clear of debris, providing paths through which fine material can pass. There is a need to reduce uncertainty in predicting the performance of these screens under a wide variety of scenarios. Since modeling everything theoretically is impractical, the emphasis should be placed on designing for predictability, supported by data.

THE PATH FORWARD

In response to GL 2004-02, licensees have undertaken the task of showing that they satisfy the requirements of recirculation core cooling. In most cases, the response has been to plan the replacement of sump screens by those with significantly larger area. The hole size and other characteristics of these screens may also be changed.

These changes are in the right direction to alleviate the potential for excessive head loss. However, in view of uncertainties introduced by new research results, the incomplete response by industry to the generic letter, the difficulties of validating the "proof tests" planned by industrial consortia, and downstream effects, NRR will need to develop assurance that it has the capability to evaluate the effects of these changes. The staff anticipates that, if sufficient uncertainty is encountered, supplemental actions may be required. These may include the following measures:

- Removal from containment of constituents that are known to cause problems with head loss and lack of predictability.

- Development of screen designs that are insensitive to the plethora of uncertainties associated with many existing designs. These designs may include active screens or similar devices that can handle many forms of debris without the need for knowing the details of the debris characteristics.

- Design of screens for minimum bypass of fine debris. Emphasis is currently being placed on reducing head loss, but downstream effects should also be considered.

- Identification of other solutions to core cooling that get around the manifold uncertainties associated with the present range of screen designs and can more confidently demonstrate success in meeting specifications.

- Use of probabilistic analysis to show that the most undesirable debris bed configurations are highly unlikely. Evaluation would be based on realistic analysis rather than on a conservative approach.

We endorse the immediate plans to increase the size of sump screens because this will alleviate the potential for excessive head loss. This action by itself may not be sufficient to resolve all long-term core cooling issues.

We anticipate working further with the staff on these important matters.

Dr. William Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,

/RA/
Graham B. Wallis
Chairman

References:

1. U.S. Nuclear Regulatory Commission Generic Letter 2004-02: "Potential Im pact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors", September 13, 2004.

2. U.S. Nuclear Regulatory Commission Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors", June 9, 2003.

3. Letter from Mario V. Bonaca, Advisory Committee on Reactor Safeguards, "Proposed Draft Final Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at PWRs," July 19, 2004.

4. "Draft NRC Staff Review G uidance for Evaluation of Downstream Effects of Debris Ingress into the PWR RCS On Long Term Core Cooling Following a LOCA", undated.

5. "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," WCAP- 16406-P, Westinghouse Owners Group, June 2005.

6. Information Notice 2005-26: "Results of Chemical Effects Head Loss Tests in a Simulated PWR Sump Pool Environment," September 16, 2005.

7. NRC Information Notice 2005-26, Supplem ent 1: "Additional Results of Chemical Effects Tests in a Simulated PWR Sump Pool Environment," January 20, 2006.

8. "Integrated Chemical Effects Test Project: Test #1 Data Report," LA-UR-05-124, June 2005

.
9. “Integrated Chemical Effects Test Project: Test #2 Data Report,” LA-UR-05-6146, September 2005.

10. “Integrated Chemical Effects Test Project: Test #3 Data Report,” LA-UR-05-6996, October 2005.

11. “Integrated Chemical Effects Test Project: Test #4 Data Report,” LA-UR-05-8735, November 2005.

12. “Integrated Chemical Effects Test Project: Test #5 Data Report,” LA-UR-05-9177, January 2006.

13. Memorandum from Michele G. Evans to John N. Hannon, “Final Transmittal of Information Summarizing Integrated Chemical Effects Results and Implications”, October 25, 2005.

14. “Corrosion Rate Measurements and Chemical Speciation of Corrosion Products Using Thermodynamic Modeling of Debris Components to Support GSI-191, ”NUREG/CR-6873, April 2005.

15. “Screen Penetration Test Report,” NUREG/CR-6885, LA-UR-04-5416, October 2005.

16. Memorandum from Ralph Architzel to James Lyons, “Report on Results of Staff Pilot Plant Audit–Crystal River Analyses Required for the Response to Generic Letter 2004-02 and GSI-191 Resolution,” June 29, 2005.

17. Memorandum from Ralph Architzel to Thomas Martin, “Report on Results of Staff Pilot Plant Audit– Fort Calhoun Station Analyses Required for the Response To Generic Letter 2004-02 and GSI-191 Resolution,” January 26, 2006.

August 1, 2006

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: GENERIC SAFETY ISSUE 191 - ASSESSMENT OF DEBRIS ACCUMULATION ON PWR SUMP PERFORMANCE

Dear Mr. Reyes:

On April 10, 2006, we issued a report to Chairman Diaz discussing the resolution of Generic Safety Issue (GSI) 191, “Assessment of Debris Accumulation on PWR Sump Performance.”

On May 2, 2006, you responded that it is the staff’s intent to terminate research activities related to GSI-191 in June 2006. You indicate that additional work by the industry and the staff may be needed to address some remaining issues such as chemical and downstream effects.

The staff’s current approach is to rely on large-scale integral tests of screens by the industry to demonstrate that the safety margin is sufficiently conservative to accommodate phenomenological uncertainties. Because of the complexity of the phenomena that affect the pressure drop across debris beds, particularly when chemical effects are included, the staff has concluded that the development of predictive models is a “challenging and long-term effort which may not achieve timely closure of GSI-191 issues.”

The efforts that are being taken by the industry in response to Generic Letter 2004-02 to substantially increase screen size are appropriate. We also agree that the industry’s integral experiments will help to support the safety case. However, it is important to recognize the limitations of these tests.

Historically, integral tests have been used to validate predictive analytical tools. These tools are used to evaluate the performance of safety systems. Integral tests have not been used as “proof tests” as an alternative to analytical tools because of the difficulty of achieving conditions that are truly prototypic. In addition, it is not practical to examine system behavior experimentally over the full range of variability of input conditions. The planned tests of full-size screen modules will be performed using conditions that vary substantially from prototypic, including differences in water temperature, water chemistry, pre-conditioning of insulation debris, and the actual system configuration, such as multiple modules. In order to understand the impact of these experimental non-typicalities, it is necessary to have some level of quantitative understanding of the phenomena. The staff must have the capability to perform an independent technical assessment of the approaches used by licensees to address GSI-191 issues.

During a meeting on June 13-14, 2006, our Thermal-Hydraulic Phenomena Subcommittee reviewed the status of the NRC’s sump performance research program. Substantial progress has been made in a number of areas. Progress on developing a predictive tool for debris bed pressure drop without chemical effects is very promising but further work is required.

Experiments have been performed that indicate that chemical effects can be substantial.

However, to date, the staff has not interpreted the experimental results from these tests within the context of a mechanistic model or even correlated them empirically. The staff only recently initiated calculations to assess potential downstream effects, particularly related to in-vessel flow blockages. These are examples of areas in which additional research is still warranted.

A continued regulatory research program to address key areas of uncertainty is a risk management strategy for reducing the likelihood of erroneous regulatory conclusions. We recommend that confirmatory research on GSI-191 be continued.

Dr. William J. Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,
/RA/
Graham B. Wallis
Chairman

References:

1. Report dated April 10, 2006 from Graham B. Wallis, Chairman, Advisory Committee on Reactor Safeguards, to Nils J. Diaz, Chairman, Nuclear Regulatory Commission, Subject: Generic Safety Issue 191 - Assessment of Debris Accumulation on PWR Sump Performance.

2. Memorandum dated May 2, 2006 from Luis Reyes, Nuclear Regulatory Commission, for Graham B. Wallis, Chairman, Advisory Committee on Reactor Safeguards, Subject: Generic Safety Issue 191 - Assessment of Debris Accumulation on PWR Sump Performance.

3. U.S. Nuclear Regulatory Commission Generic Letter 2004-02: "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors," September 13, 2004.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
556TH meeting

SELECTED CHAPTERS OF THE SAFETY EVALUATION REPORT (SER) ASSOCIATED WITH THE ECONOMIC SIMPLIFIED BOILING WATER REACTOR (ESBWR) DESIGN CERTIFICATION APPLICATION

Thursday, October 2, 2008

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1.	Opening Remarks	M. Corradini	1:15 pm – 1:25 pm
2.	GEH presentation of Chapter 19 & 22 of the DCD	Jeff Waal, GEH	1:25 pm – 2:15 pm
3.	Staff presentation of Chapter 19 & 22 Of the ESBWR SER with Open Items	R. Foster, NRO	2:15 pm - 3:15 pm

Presentation time should not exceed 50% of the total time allocated for a specific item. Number of copies of presentation material to be provided to the ACRS – 35

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

……………………………………………..

Cognizant ACRS Member: Michael Corradini
Cognizant ACRS Staff Engineer Harold VanderMolen

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
ESBWR DESIGN CERTIFICATION
OCTOBER 2, 2008
ROCKVILLE, MARYLAND

- STATUS REPORT -

PURPOSE

The purpose of this meeting is to review Chapters 19 and 22 of the SER with Open Items for the ESBWR Design Certification (Refs.1 and 2). GE-Hitachi Nuclear Americas LLC (GEH), the applicant for this design, and the Office of New Reactors (NRO) staff will provide presentations regarding the review of the ESBWR Design Certification for the topics in this chapter. The ESBWR Subcommittee met on June 3, 2008, to review these chapters of the SER with Open Items and provided several comments to the staff and GEH for their consideration. The ESBWR Subcommittee met again on August 21 and 22, 2008, to review the probabilistic risk analysis (PRA) (Ref. 3) supporting Chapter 19, and to continue the review of Section 19.2, on severe accident mitigation.

BACKGROUND

The ESBWR has a different design and different operating characteristics compared to currently operating BWRs. The design uses natural circulation and passive ECCS, which requires a reactor vessel with a relatively large steam volume at power and a relatively large water volume when shutdown. The reactor is designed to provide a more gentle response to design basis transients and accidents. The design uses passive containment cooling together with passive drywell flooding. A lower drywell core catcher is included in the design for severe accidents.

A complete description of the ESBWR design and safety features is provided in the ESBWR Design Control Document.

DISCUSSION

GEH submitted the ESBWR design certification application on August 24, 2005. Subsequently, the staff identified deficiencies in the application and GEH submitted additional material. The staff formally accepted the complete application on December 1, 2005. During the review, the staff issued approximately 3100 Requests for Additional Information (RAIs) and prepared SER inputs with open items identified. The staff is providing the SERs with Open Items to the ACRS for review chapter-by-chapter.

A summary of the SER with Open Items for Chapters 19 and 22 is provided below. Comments and critiques by the ACRS staff are printed in italics.

CHAPTER 19. PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENTS

DISCUSSION

PROBABILISTIC RISK ASSESSMENT (19.1)

The purpose of the staff's review is to ensure that the applicant has adequately addressed the objectives established by the Commission. These objectives, which include the main uses of any PRA, include the following:

- Use the PRA to perform the following:
  - Identify and address potential design features and plant operational vulnerabilities, where a small number of failures could lead to core damage, containment failure, or large releases (e.g., assumed individual or common-cause failures (CCFs) could drive plant risk to unacceptable levels with respect to the Commission's goals, as presented below).
  - Reduce or eliminate the significant risk contributors of existing operating plants that are applicable to the new design, by introducing appropriate features and requirements.
  - Select among alternative features, operational strategies, and design options.
- Identify risk-informed safety insights based on systematic evaluations of the risk associated with the design such that the applicant can identify and describe the following:
  - the design's robustness, levels of defense-in-depth, and tolerance of severe accidents initiated by either internal or external events
  - the risk significance of potential human errors associated with the design
- Determine how the risk associated with the design compares against the Commission's goals of less than $1x10^{-4}$/yr for core damage frequency (CDF) and less than $1x10^{-6}$/yr for large release frequency (LRF). In addition, compare the design against the Commission's approved use of a containment performance goal (CPG), which includes (1) a deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges and (2) a probabilistic goal that the conditional core damage probability (CCDP) be less than 0.1 for the composite of all core damage sequences assessed in the PRA.
- Assess the balance between features of the design that prevent or mitigate severe accidents.
- Determine whether the plant design represents a reduction in risk compared to existing operating plants.[1]

- Demonstrate compliance with 10 CFR 50.34(f)(1)(i), which requires that a plant-specific PRA be performed to seek improvements in the reliability of core and containment heat removal (CHR) systems that are significant and practical.

- Use the PRA in support of the process employed to determine whether regulatory treatment of non-safety systems (RTNSS) is necessary and, if appropriate, the systems, structures, and components (SSCs) included in RTNSS.

- Use the PRA in support of programs associated with plant operations (e.g., technical specifications, reliability assurance, human factors, and maintenance).

- Use the PRA to identify and support the development of specifications and performance objectives for the plant design, construction, inspection, and operation, such as inspections, tests, analyses, and acceptance criteria (ITAAC), reliability assurance program (RAP), technical specifications (TSs), and combined license (COL) action items and interface requirements.

Historically, the staff reviews of industry-generated PRAs have consisted of two parts, a review of the quality of the PRA, and then the use of that PRA to generate insights into the overall risk profile of the plant. There are many uses for this knowledge – the designer can identify and address the points where the design is most vulnerable, and the regulator can identify where vigilance is called for.

QUALITY OF THE PROBABILISTIC RISK ASSESSMENT (19.1.2)

The ESBWR PRA is a full-scope (Levels 1, 2, and 3) PRA that covers both internal and external events for at-power and shutdown operations.

The methodology used in the ESBWR Level 1 PRA is a (well-established) linked fault tree approach. Fault trees have been developed and evaluated for the major ESBWR front-line and support systems to determine the probability that emergency core cooling and decay heat removal (DHR) systems perform their intended function when demanded. Transient and loss-of-coolant accident (LOCA) initiating events have been consolidated into major accident event sequences that are described by the accident event trees. These event trees are used to calculate the frequency of core damage sequences by directly linking the fault trees and solving for the minimal cutsets. Outcomes of the event trees are transferred to containment event trees (CETs) for further treatment to determine frequencies of radioactive releases to the environment.

Results of the CET analyses provide the necessary input to model and assess the transport of fission products through the drywell and containment, calculate fission product release fractions associated with containment release paths, and determine potential consequences associated with each fission product release category.

Initiating event frequencies are based on generic industry data from existing BWR operating experience. Component failure probabilities were estimated based on generic industry data and on ESBWR design-specific information. The human error probabilities used in the model are conservative screening values extracted from industry and NRC publications.

Severe accident phenomena are explicitly addressed and are quantitatively treated. The Risk-Oriented Accident Analysis Methodology (ROAAM) is used to assess the containment response to severe accident phenomena. A linked fault tree approach is used to address the containment systems and the ability to prevent overpressurization from loss of decay heat removal.

To support the consequence analysis, multiple radionuclide release categories are modeled. Source terms are defined based on ESBWR thermal-hydraulic (T-H) analyses. Bounding consequence analyses are performed, showing that the ESBWR design meets NRC safety goals with sufficient margin.

The external events portion of the PRA explicitly analyzes core damage accidents initiated during power and shutdown operation for the following hazards:

- internal floods
- internal fires
- high winds
- seismic events

The external events analyses are bounding assessments that are meant to show significant design margin for these hazards. The frequencies of initiating events are based on generic industry data and are applied in a bounding manner. The fault trees and event trees developed for the internal events evaluations are used in the external events analyses to the maximum extent possible, using logic flags that account for the common failures induced by the external hazard events. The ESBWR seismic assessment is a seismic margin analysis (SMA). The analysis demonstrates that the ESBWR plant and equipment can withstand an earthquake with a magnitude at least 1.67 times that of the safe-shutdown earthquake (SSE).

The staff reviewed the quality of the ESBWR PRA (including NEDO-33201,“ESBWR Probabilistic Risk Assessment,” Revision 2, issued September 2007, hereafter referred to as the PRA report) by conducting its own independent evaluation of the applicant’s use of models, techniques, methodologies, assumptions, data, and computational tools, as well as evaluating the applicant’s programs and processes for ensuring quality in the PRA. The staff’s review covered all aspects of the PRA model and the use of the model to assess the ESBWR, including assumptions, data, modeling, quantification, uncertainties, and sensitivity studies. The applicant has responded to the majority of the staff’s requests for additional information (RAIs), and the staff has found the responses to be acceptable. The applicant has incorporated information provided in these RAI responses into Revision 2 of the PRA report and Revision 4 of the DCD, as appropriate.

PRA Technical Adequacy

The staff also considered the extent to which the applicant’s PRA conforms to existing consensus standards for PRA (American Society of Mechanical Engineers (ASME)-RA-Sb-2005, “Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications” that have been endorsed by the NRC staff. The applicant has stated that, “Where applicable, ASME-RA-Sb-2005 capability Category 2 attributes are included in the analysis.” The staff has asked for information regarding specifically which attributes were <u>not</u> included. This is being tracked as an open item, pending a satisfactory reply.

PRA Maintenance Program

The staff has reviewed the applicant's proposed maintenance and update program and determined that the program includes the key elements described in RG 1.200. The program described by the applicant in the DCD is therefore acceptable.

SPECIAL DESIGN FEATURES (19.1.3)

The ESBWR design, compared with older BWR designs, includes a number of new features aimed at preventing core damage. The staff SER summarizes them as follows:

- For prevention and mitigation of an anticipated transient without scram (ATWS), the ESBWR is designed with the following features:

  - an alternate rod insertion (ARI) system that utilizes sensors and logic that are diverse and independent of the reactor protection system (RPS)

  - electrical insertion of fine motion control rod drives (FMCRDs) that also utilize sensors and logic that are diverse and independent of the RPS

  - automatic feedwater runback under conditions indicative of an ATWS

  - automatic initiation of SLCS under conditions indicative of an ATWS

  - elimination of the scram discharge volume in the control rod drive system (CRDS)

  DCD Section 15.5.4 provides details on the effectiveness of these design features for addressing ATWS concerns. Given these features, ATWS contributes insignificantly to CDF and LRF, as shown in the ESBWR PRA.

- The design of the ESBWR reduces the possibility of an intersystem loss-of-coolant accident (ISLOCA) outside containment by designing to the extent practicable all piping systems, major system components (pumps and valves), and subsystems connected to the reactor coolant pressure boundary (RCPB) to an ultimate rupture strength at least equal to the full RCPB pressure. Because of these design features of the ESBWR, ISLOCA is not a significant contributor to initiating events or accidents.

- The ESBWR design reduces the frequency and consequences of LOCAs resulting from large-diameter piping failure by removing the recirculation system altogether.

- The ICS consists of four totally independent trains, each containing an isolation condenser (IC) that condenses steam on the tube side and transfers heat to the isolation condenser/passive containment cooling system (IC/PCCS) pool, which is vented to the atmosphere. The ICs, which are connected by piping to the reactor pressure vessel (RPV), are placed at an elevation above the source of steam (i.e., vessel). When the steam is condensed, the condensate is returned to the vessel via a condensate return line. The ICS is designed as a safety-related system to remove reactor decay heat following reactor shutdown and isolation in a passive way and with minimal loss of

coolant inventory from the reactor, when the normal heat removal system is unavailable following any of the following events:

– sudden reactor isolation from power operating conditions
– station blackout (SBO) (unavailability of all alternating current (ac) power)
– ATWS
– LOCA

The ICS also prevents unnecessary reactor depressurization and operation of other engineered safety features that can also perform this function. In the event of a LOCA, the ICS provides additional liquid inventory from an inline condensate reservoir upon opening of the condensate return valves to initiate the system.

- The GDCS provides emergency core cooling passively after any event that threatens the reactor coolant inventory. Once the nuclear boiler system (NBS) has been depressurized via the automatic depressurized system (ADS), the GDCS is capable of passively injecting large volumes of water into the depressurized RPV to keep the fuel covered over both short and long timeframes following system initiation.

- The fuel and auxiliary pools cooling system (FAPCS) is designated as a backup system for low-pressure coolant injection (LPCI). In LPCI mode, the system provides makeup water from the suppression pool to the RPV through one of the main feedwater lines after the reactor has been sufficiently depressurized. The FAPCS can also provide backup shutdown cooling water. The FAPCS can provide cooling water during the long term using a pipe connection to convey water to the ICS/PCCS pool for post-LOCA heat removal after 72 hours.

- During a total loss of offsite power, the safety-related electrical distribution system is automatically powered from the onsite, non-safety-related diesel generators. If, however, these diesel generators are not available, each division of the safety-related system independently isolates itself from the non-safety-related system, and the safety-related batteries of each division provide uninterrupted power to safety-related loads of each safety-related load division. The divisional batteries are sized to provide power to required loads for 72 hours. In addition, devices that monitor the input voltage and frequency from the non-safety system and isolate the division automatically on degraded conditions protect each division of the safety-related system. The combination of these factors in the design minimizes the probability of losing electric power from onsite power supplies as a result of the loss of power from the transmission system or any disturbance of the non-safety-related ac system. Because of the nature of the passive safety-related systems in the ESBWR, SBO events are not significant contributors to CDF or LRF.

- The PCCS is a safety-related, passive-acting CHR system that maintains the containment within its design pressure and design temperature limits for design-basis accidents (DBAs) including LOCAs and post-blowdown events. The PCCS also provides a flowpath for released steam vapor back to the RPV through the GDCS. Because the PCCS is highly reliable as a result of its redundant heat exchangers and totally passive component design, the probability of a loss of CHR is significantly reduced.

- The fire protection system (FPS) serves as a preventive feature for severe accidents in two ways. First, it reduces or eliminates the possibility of damaging fire events that could induce transients, damage mitigation equipment, and hamper operator responses. Second, it supplies a means for long-term makeup to the upper containment pools, which may be required after the first 72 hours of an accident requiring passive heat removal.

These new features greatly reduce or eliminate many of the accident sequences that are found in the PRAs of the existing fleet of BWRs. This is highly desirable, of course, but a consequence of eliminating these sequences is to make a probabilistic analysis more difficult.

The analysis also describes a number of features which, if a severe damage event should occur, mitigate the accident progression and prevent releases of radioactivity to the environment. Some of these are:

- The ESBWR containment is designed to a higher ultimate pressure.
- The containment atmosphere is inerted to prevent any combustible gas deflagration (a feature that was used in the BWR/2 through BWR/4 product line, but not in the more modern plants that use a Mark II or Mark III containment).
- The number of penetrations is minimized.
- The deluge mode of the GDCS will flood the lower drywell to cover any molten core debris.
- The BiMAC device will passively cool core debris on the drywell floor.

SAFETY INSIGHTS FROM THE INTERNAL EVENTS PRA FOR OPERATIONS AT POWER (19.1.4)

The applicant reports a total CDF resulting from internally generated accident sequences during power operations of 1.22×10^{-8}/yr.

- Dominant sequences typically do not contain independent component failures. Instead, they consist of CCFs that disable entire mitigating functions. It is important to note that multiple mitigating functions must fail in the dominant sequences. A single common-cause event is not sufficient to directly result in core damage.

- The ESBWR Level 1 PRA CDF is significantly impacted if the non-safety-related systems are not credited. If the analysis takes credit for all the key backup non-safety systems, the focused Level 1 PRA results are reduced by almost 2 orders of magnitude. However, the impact to the CDF can be minimized by about an order of magnitude if the analysis credits only the availability of the DPS (including surrogate logic for DPS signal for main steam isolation valve (MSIV) isolation). *(This will be discussed at more length in Chapter 22.)*

- ATWS events are low contributors to plant CDF because of the improved scram function and passive boron injection.

- In core damage sequences involving failure of the ICS where high-pressure makeup has failed and either failure to depressurize occurs or low-pressure injection is not available, the failure of the PCCS or the failure to provide makeup to the pools is not a significant contributor to CDF.

Transients contribute the most to CDF, approximately 85 percent. The most significant groups of transient initiators are as follows:

- inadvertent stuck-open relief valve (36.5 percent)
- general transients (20.4 percent)
- loss of feedwater transients (16.7 percent)
- loss of offsite power transients (11.5 percent)

LOCAs that occur inside containment contribute approximately 9 percent. The most significant LOCA initiator with respect to CDF contribution is the large steam break in feedwater line B, which represents about 4 percent of the overall CDF, thus becoming the fifth most important initiating event. Finally, breaks outside containment represent less than 3 percent of the total value of the CDF.

To put this in some perspective, the NUREG-1150 project estimated CDFs for Peach Bottom (a BWR/4 in a Mark I containment) and Grand Gulf (a BWR/6 in a Mark III containment) to be in the 10^{-6} to 10^{-5} range, which is considered low compared to many other current plants. (The staff's write-up also compares the ESBWR CDF to those presented in the IPE program.) A CDF in the 10^{-8} range is extremely low, but not surprising if the designers systematically use the PRAs of existing plants to design away the dominant accident sequences of these plants. The reader should remember that PRAs only model failure modes that are already known to the analyst, and failures yet to be discovered are obviously not included.

Nevertheless, even with this caveat, a low CDF estimate such as this is indeed a desirable outcome.

Moreover, the spread of contributors over a variety of initiating events, with no single group of sequences dominating, is a desirable outcome. Another caveat: these sequence frequencies are generally cast on logarithmic scales and have large uncertainties. Percentage breakdowns (such as 9% due to LOCAs) are not very robust – slight changes in some of the initiating event frequencies or failure probabilities can completely change the picture painted by these percentages.

The staff SER lists the following key insights:

- Sensitivity study results indicate that changes in the human error failure probabilities, particularly preinitiators, have the potential to impact CDF.

- Sensitivity study results indicate that squib valve failure rate estimates have the potential to impact CDF.

- Sensitivity study results indicate that the preinitiator operator actions have a significant impact on the risk achievement worth (RAW) value. This is primarily because of the many potential latent failures and relatively high reliability estimate for each of these operator actions.

- Accident sequences in which DPVs are challenged contribute to approximately 61 percent of the CDF. In two-thirds of the cases, the DPVs are demanded and are successful; in one-third of the cases, the DPVs are demanded and fail.

- The PRA model conservatively assumes that a single failure on either train of the SLCS causes core damage if the control rods fail to insert (ATWS). The CDF would be reduced by approximately 13 percent if either train of the SLCS is able to mitigate ATWS scenarios.

- Changes to the squib valve failure rate have a significant impact on CDF. Increases in the failure rate of the squib valves used for the ADS cause a significant increase in the accident Class III contribution (core damage at high reactor vessel pressure). Similarly, an increase in the failure rate of the squib valves in the GDCS causes a significant increase in the accident Class I contribution (core damage at low reactor vessel pressure). However, an increase only to the SLCS squib valves does not have a very pronounced impact on CDF.

- An increase of 1 order of magnitude of the vacuum breaker and backup valve failure rate causes the CDF to increase only by approximately 10 percent.

The staff concluded that the applicant has performed adequate systematic evaluations of the risk associated with the design and used them to identify risk-informed safety insights in a manner consistent with the Commission's stated goals.

SAFETY INSIGHTS FROM THE LEVEL 2 INTERNAL EVENTS PRA (CONTAINMENT ANALYSIS)

The ESBWR has a very low large release frequency (LRF) ($9.6x10^{-10}$/yr), and accident sequences leading to such results are not only unlikely but also have broad bands of uncertainties associated with them. Consequently, the applicant used a bounding approach, rather than a best-estimate method, for assessing containment performance. The applicant also estimated that the ESBWR passive containment design is sufficiently robust to effectively mitigate the consequences of severe accidents with a low attendant CCFP of 0.08.

The applicant's Level 2 analysis uses containment event trees in a fairly standard approach. The methodology used includes binning the Level 1 PRA results into a manageable number of accident classes and constructing and quantifying containment event trees (CETs), simulating severe accident progression and containment challenges for a number of accident sequences that represent the significant core damage scenarios, and assigning representative sequence results into release categories for the purpose of defining the end states and determining the pathways of radioisotopes into the environment.

This technique has been in use since the 1990s. The part of the level II analysis that is generally of interest is not so much the containment event tree logic, but the phenomenological effects that are used to calculate the split fractions.

The applicant's Level 2 analysis contained a number of points of interest. Some of the more interesting include the following:

- The applicant concluded that an ex-vessel steam explosion from molten core material dropping into a deep pool of water was physically unreasonable. *This was the source of considerable dialog in the review of the ABWR design in the 1990s, where the concern*

was that the drywell might be flooded before, not after, the failure of the lower reactor vessel head. The applicant argued, among other things, that the physical fact that premixtures in saturated water pools become highly voided and thus unable to support the escalation of natural triggers to thermal detonations, and also that the reactor pedestal and BiMAC structural designs are capable of resisting explosion load impulses. Similar calculations performed under NRC sponsorship confirmed the applicant's conclusions

- Because of the ESBWR design and reliability of containment systems, the most likely containment response to a severe accident is associated with successful containment isolation, successful vapor suppression, and successful CHR. As a result, the containment provides a highly reliable barrier to the release of fission products after a severe accident, with only 8 percent of the core damage accidents resulting in releases larger than those associated with the minimal release leakage at the TS limit. This result meets the recommended goal of 10 percent.

- A containment penetration screening evaluation indicated that only a few penetrations required isolation to prevent significant offsite consequences. The probability of the bypass failure mode is dominated by common-cause hardware failures, resulting in a calculated frequency of containment bypass about 3 orders of magnitude lower than the TSL release category.

In its evaluation, the staff noted the following:

The staff's review of Chapter 19 of the ESBWR DCD and Sections 8–11 and 21 of the PRA verifies that the ESBWR design is more robust and has greater tolerance for severe accidents than that of the operating plants. Specific findings include the following:

- The LRF for internal events is calculated by the applicant to be 9.6×10^{-10}/yr, and the CCFP is calculated to be 0.08. The LRF is more than 3 orders of magnitude below the Commission's safety goal, and the CCFP is acceptably low. This is a significant reduction in risk as compared with existing BWRs, which typically have LRF values in the range of 1.0×10^{-6}/yr to 1.0×10^{-5}/yr and CCFPs up to 0.7, with an average value around 0.3.

- The design features and requirements introduced by the applicant reduce or eliminate significant risk contributors identified in existing operating plants. These features provide a good balance between prevention and mitigation.

  – The new features designed to prevent or mitigate ATWS greatly reduce the probability and/or consequences of ATWS and hence LRF.

  – Designing all piping systems, pumps, valves, and subsystems connected to the RCPB to an ultimate strength equal to or greater than the full RCPB pressure is a preventive measure that reduces the likelihood of ISLOCA and consequent containment bypass probability and hence LRF.

  – Since the ESBWR containment is designed to a higher ultimate pressure than that of currently operating BWRs, there is a higher likelihood of averting containment failure and hence a reduction in LRF and CCFP. The containment

would be more likely to survive for at least 24 hours following the onset of core damage.

– The probability of a high-pressure core melt is reduced as a result of a highly reliable ADS. This system plays a role both in preventing and mitigating severe accidents. It reduces the likelihood of early containment failure from DCH. Moreover, drywell segregation into upper and lower regions, and the ability to vent the UDW atmosphere into the wetwell through a large venting area, would mitigate the effects of a high-pressure core melt. Consequently, the risk impacts of high-pressure core melt events (LRF and CCFP) are reduced in comparison to those of current-generation BWRs.

– The deluge mode of GDCS operation, in concert with the BiMAC device, would act to further reduce the likelihood of containment failure, either from overpressurization, drywell liner melt-through, or from basemat penetration from core debris attack. Moreover, the design procedure of not immediately adding water greatly reduces the probability of a highly energetic steam explosion. Consequently, LRF and CCFP are further reduced relative to current-generation BWRs.

– The wetwell vent is available to avert catastrophic containment failure. It would not be needed during the first 24 hours after core damage and would be opened only if the containment pressure exceeded 90 percent of its ultimate capacity.

The staff also performed a number of audit calculations to verify these statements.

RESULTS AND INSIGHTS FROM THE LEVEL 3 INTERNAL EVENTS PRA

The applicant’s Level 3 analysis used the MACCS2 code to calculate the consequences of each release category. These were then multiplied by the frequencies of these release categories, and the products summed to calculate the risk factors.

The applicant performed the MACCS2 calculation using a meteorological condition comparable to the EPRI ALWR URD meteorological reference data set, which is indicated to be a meteorological data set significantly worse than conditions at the average U.S. site. The SANDIA siting study population density data are used to develop a uniform population density. A bounding uniform density of 305 people per km^2 (790 people per mi^2) for the first 32 kilometers (20 miles) is used for all radial intervals. The evacuation parameters used in this analysis are termed conservative assumptions in that no evacuation or relocation in terms of physical movement is assumed and no sheltering is assumed. The public is assumed to continue normal activity during the reactor accident in this bounding analysis. For baseline analysis, each release category and associated source term is modeled to occur at ground level. The thermal content of the plume is assumed to be the same as ambient.

The staff found the overall approach to consequence analysis and the use of the MACCS2 code to be consistent with the present state of knowledge regarding severe accident modeling and therefore acceptable. However, the calculation included only the internal-events portion of the PRA. The lack of a Level 3 analysis for external events or shutdown modes is being tracked as an open item.

The insights derived from this portion of the analysis are summarized as follows:

- The estimated total risk to the public for the ESBWR design is low and acceptable. Offsite risk is very low compared to that of the current generation of operating plants because of a combination of (1) a very low estimated CDF, (2) a low CCFP, and (3) a relatively benign source term associated with the frequency-dominant release category.

- The risk results demonstrate that the ESBWR, for accidents arising from internal events during full-power operation, meets the established consequence-related goals with substantial margin.

- The results for the ESBWR do not explicitly include the contribution to risk from external events. The surrogate risk results for externally initiated events and shutdown operations give confidence that the ESBWR would still meet the Commission's safety goal policy with margin when these additional contributors are included.

- The release category associated with normal containment leakage levels is a low but not negligible contributor to the public risk. It is assigned to every core damage accident.

- The containment failure accident release categories contributing most to the public risk (ex-vessel explosions (EVE), breaks outside containment (BOC), and containment bypass (BYP)) have conditional probabilities of occurrence of 0.05 or less. For EVE, this results primarily from the design-driven low probability of high levels of water being present in the LDW just before vessel failure; for BOC, designing to the extent practical all components connected to the RCPB to an ultimate rupture strength at least equal to the full RCPB pressure; and for BYP, the minimization of the number of penetrations.

- The other containment failure accident release categories contributing to the public risk have conditional probabilities of occurrence of 0.01 or less. These low probabilities are largely attributable to the presence of the BiMAC device.

The applicant has conservatively chosen to designate all containment failures as large release. The staff found this to be acceptable. The class of severe accidents in which the only release from the containment is at the design leakage rate level is not included in this designation, but it is included in the risk calculations.

SAFETY INSIGHTS FROM THE EXTERNAL EVENTS PRA FOR OPERATIONS AT POWER (19.1.5)

Seismic Risk Assessment

The seismic risk assessment is a seismic margins analysis (SMA), which is based on probabilistic analysis techniques, but is not a true PRA. This approach has been used previously in the IPEEE program and in other design certifications. Unlike a seismic PRA, a seismic margins analysis does not result in an estimate of core damage frequency from seismic events. Instead, it is a method for estimating how severe an earthquake must be before plant safety is compromised.

The ESBWR is designed to withstand a safe shutdown earthquake (SSE) of 0.5g. The NRC has indicated in SECY-93-087 and the associated SRM that a plant designed to withstand a 0.5-g SSE should have a plant HCLPF capacity of at least 1.67 times the acceleration of the

SSE (in this case, 0.84 g). The applicant concluded, based on this analysis, that the ESBWR could indeed withstand a ground acceleration of this magnitude, and the staff found this to be acceptable.

Internal Fire Risk Assessment

A fire probabilistic risk assessment (FPRA) is performed taking into account that the specifics of cable routings, ignition sources, and target locations in each zone of the plant are not known at this stage of the plant design. Because of this limitation, the applicant used a simplified conservative and bounding approach. For example, the FPRA assumes the worst effects of fire on all the equipment and systems located in each group of fire areas; that is, any fire in any fire area will cause the worst damage, and a fire ignition in any fire area continues to grow unchecked into a fully developed fire without credit for fire suppression.

The total CDF for fire events at full power is 8.06×10^{-9}/yr. The total LRF for fire events at full power is 4.83×10^{-10}/yr. The staff did have some trouble duplicating some of the applicant's calculations, and currently has an open item on this subject.

The fire analysis is fairly standard, and the approach used is not greatly affected by the unique features of the ESBWR design. The resulting risk numbers are very low. However, experience with existing designs has taught the staff that a certain amount of vigilance is needed to make sure that these numbers are low. Routine plant operations and maintenance must be such that the assumptions made in an analysis such as this are valid.

Internal Flooding Analysis

Floods may be caused by large leaks resulting from the rupture or cracking of pipes, piping components, or water containers such as storage tanks. Another possible flooding cause is the operation of fire protection equipment.

A flooding event may result in an initiating event and may also disable mitigating systems. Thus, buildings containing mitigating equipment credited in the PRA accident sequence analysis, or equipment whose loss could cause an initiating event, are of interest in the flooding analysis.

The staff accepted the applicant's analysis, which is fairly standard. The total CDF for full-power internal flooding events was 1.62×10^{-9}/yr. The total release frequency for internal flooding events excluding TSL at full power was 2.7×10^{-10}/yr.

SEVERE ACCIDENT EVALUATIONS (19.2)

This rather voluminous section evaluates the ability of the ESBWR to cope with a severe accident (i.e., a core melt). Although the ESBWR has a number of features which reduce the likelihood of such and event to a very low value, the design also incorporates a number of features which help prevent a significant release of radioactivity should such an accident occur. These features include the following:

- The containment atmosphere is inerted during operation, precluding any hydrogen combustion events.
- The containment basemat incorporates the BiMAC (Basemat Internal Melt Arrest and Coolability) device. This device consists of a series of inclined pipes under a refractory

material (ceramic zirconia), all located on the lower drywell floor. The pipes are filled with water from the GDCS pools by means of squib valves, and are capable of providing sufficient cooling by natural circulation to prevent basemat melt-through.
- If molten core material is present on the drywell floor, the lower drywell deluge system, in addition to supplying water to the BiMAC device, will cover the molten core material with a layer of water. (There is a diverse detection and activation system to ensure that the water arrives after the core material pool forms.)
- There is a water pool located above the upper drywell head, which normally functions as a radiation shield, but also provides cooling to the drywell head itself, which is immersed in the pool water.
- In-vessel steam explosions due to fuel-coolant interaction as molten material drops into water in the lower vessel plenum are prevented by the lower plenum design, which is densely occupied by control rod guide tubes.
- Ex-vessel explosions are rendered unlikely by the accident management strategy that prevents a deep pool of water from forming before the molten material falls into the lower drywell area. In addition, the reactor pedestal and BiMAC device are capable of resisting high impulse loads from explosions. Finally, such a pool is likely to be highly voided and unlikely to generate a large detonation-like event.
- The containment structure is designed to withstand a higher internal pressure (45 psig at 340°F) than are the current generation of BWR containments.

Most of Section 19.2 of the SER is a review of the capability of these features to keep a molten core from causing a large release of radioactivity into the environment. Some highlights of this review are as follows:

- The staff reviewed the BiMAC device in some detail, and found that it was based on sound analytical considerations. However, the staff noted that the applicant had carried out a testing program to demonstrate that the BiMAC device would effectively remove the decay heat in the core debris and thus confirm the design. The staff has requested documentation of the test results. This is being tracked as an open item.

- Although containment bypass events contribute a relatively small fraction of the overall risk figures, the staff has several open items on vacuum breaker performance.

- The ESBWR design does not credit the use of containment venting to prevent containment failure. The analysis includes containment venting simply to mitigate the magnitude of radionuclide releases resulting from loss of containment heat removal by forcing the pathway through the suppression pool.

Regarding the first item, concerning the testing program for the BiMAC device, GEH submitted topical report NEDE-33392P, “The MAC Experiments: Fine Tuning of the BiMAC Design,” in March of 2008 (Ref. 4). The report describes a set of scaled experiments using electrical heaters to simulate the thermal loads of a molten core. The report (still under review) claims to demonstrate significant margins to failure. (The report was presented and discussed at the August 21-22 subcommittee meeting.)

In its conclusion, the staff noted that the applicant made extensive use of the results of the PRA to arrive at a final ESBWR design. As a result, the estimated CDF and risk calculated for the ESBWR design are very low. The low CDF and risk for the ESBWR design are a reflection of the applicant’s efforts to systematically minimize the effect of initiators/sequences that have

been important contributors to CDF in previous BWR PRAs. This minimization has been done largely through the incorporation of a number of hardware improvements in the ESBWR design. Section 19.1 of this report discusses these improvements and the additional ESBWR design features that contribute to low CDF and risk for the ESBWR.

Because the ESBWR design already contains numerous plant features oriented toward reducing CDF and risk, the benefits and risk reduction potential of additional plant improvements is significantly reduced. This reduction is true for both internally and externally initiated events. Moreover, with the features already incorporated in the ESBWR design, the ability to estimate CDF and risk approaches the limitations of probabilistic techniques.

However, due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

CHAPTER 22. REGULATORY TREATMENT OF NON-SAFETY SYSTEMS

The ESBWR is designed to rely upon passive systems for safety purposes. Consequently, except for some instrumentation and control systems, active systems (systems that require AC power to operate) are designated as non-safety.

Nevertheless, some of these active systems can also mitigate transients and accidents, and thus provide additional defense in depth for the ESBWR. Because of this, these systems may be subjected to some regulatory attention, even though the various transient and accident analyses may not take credit for these systems. (Such systems are referred to as "safety-related.") Moreover, such systems may help protect a licensee's investment by precluding challenges to the safety systems.

REGULATORY TREATMENT OF NON-SAFETY SYSTEMS PROCESS (22.2)

The Regulatory Treatment of Non-Safety Systems (RTNSS) process applies to these "non-safety" systems which perform risk-significant functions. The RTNSS process uses five criteria to determine if a system should receive regulatory oversight.

1. The system is relied upon to meet deterministic NRC performance requirements.
2. The system is relied upon to ensure long-term safety (beyond 72 hours) or to address seismic events.
3. Credit for the system is needed to meet the NRC's safety goal guidelines
4. The system is needed to meet the containment performance goal, including containment bypass, during postulated severe accidents.
5. The system is needed to prevent significant adverse systems interactions.

The specific steps were established by the staff in Reg Guide 1.206.

The first step in this process is to perform a comprehensive Level III baseline probabilistic risk analysis, including all appropriate internal and external events for both power and shutdown operations. (A margins approach was used for seismic events.)

A search for adverse systems interactions between the active and passive systems is then performed. This step apparently is used to initiate design improvements to minimize or remove the interactions, not just to add the affected systems to the RTNSS list.

Then, a focused probabilistic risk assessment is performed, in which the passive systems and a minimum set of the candidate active systems were modeled. In this way, those active systems, if any, needed to meet the safety goal guidelines of a CDF of less than 10^{-4} per reactor-year and a large release frequency of less than 10^{-6} per reactor-year can be identified.

Lastly, the important non-safety-related systems are selected, using the five criteria listed above.

Once these selection steps are completed, the applicant is to determine and document the functional reliability and availability missions of these systems, and propose regulatory oversights.

APPLICANT'S EVALUATION OF SYSTEMS FOR INCLUSION IN THE RTNSS PROCESS

The baseline probabilistic risk analysis is covered in Chapter 19. The analysis included an evaluation of potential uncertainties associated with the assumptions made in the PRA modes of the passive systems.

The uncertainty evaluation determines which structures, systems and components (SCCs) are important to add margin to compensate for the PRA uncertainties. Two SCCs were identified:

- The low pressure core injection capability of the Fuel and Auxiliary Pools Cooling System (FAPCS), including support systems, was added to the ESBWR design. This capability provides a diverse backup for the passive GDCS core injection function.
- The Basemat internal Melt Arrest and Coolability (BiMAC) device was added to the ESBWR design. This device provides devense in depth protection against containment failure.

This PRA included five at-power initiating event categories and three shutdown event categories:

- Generic transients
- Inadvertent opening of a relief valve
- Transient with loss of feedwater
- Loss of preferred power
- Loss of coolant accident (LOCA)
- Shutdown loss of decay heat removal
- Shutdown loss of offsite power
- Shutdown LOCA

The applicant found the following events to be significant contributors to CDF and large release frequency:

- The loss of preferred power event, for both at-power and shutdown events. However, the dominant risk contributions are from loss of incoming AC power due to grid and weather-related faults, which are not under the site organization's control.
- The loss of feedwater event. Therefore, the condensate and feedwater systems are RTNSS candidates. However, instead of proposing availability controls, the applicant added several features to the feedwater level control system to improve its reliability.

The NRO staff reviewed the applicant's approach and findings, and found them to be acceptable, except for an assumption that both trains of the RWCU/SDC would be running. (See Chapter 19, Section 19.1.6.1.5.2 of the staff SER.)

CONTAINMENT PERFORMANCE CONSIDERATION

The containment performance goal is that the containment should maintain its role as a reliable, leak-tight barrier by ensuring that containment stresses do not exceed ASME service level C limits for a minimum period of 24 hours following the onset of core damage, and that following this 24-hour period the containment should continue to provide a barrier against the uncontrolled release of fission products.

The applicant has assessed compliance of the ESBWR design with the probabilistic containment performance goal of 0.1 CCFP with and without credit for non-safety-related SSCs,

and asserts that the goals for CDF and large release frequency can be met by crediting the Diverse Protection System (DPS) and portions of the Alternate rod Insertion (ARI) feature.

The applicant has also addressed the potential for steam bypass of the suppression pool and potential failure of the passive containment cooling system (PCCS) heat exchanger tubes in the design of the ESBWR.

The staff's evaluation of the deterministic containment performance assessment, Level 2 focused PRA, and PCCS heat exchanger tube design appear in Sections 19.2.4.3, 19.1.7.4.3.3, and 6.2 of the staff SER. These staff evaluations contain open items which must be resolved before it can conclude that no additional non-safety-related systems are needed to address concerns related to these design features.

Thus, the staff was unable to finalize its conclusions regarding acceptability.

SEISMIC CONSIDERATIONS

Based on the seismic margins analysis, the applicant indicated that no accident sequence has a high confidence of low probability of failure (HCPLF) ratio less than 1.67 times the peak ground acceleration of the safe-shutdown earthquake (SSE). Therefore, no additional non-safety-related SSCs are identified as RTNSS candidates because of seismic events.

The staff did ask some questions to verify the HCPLF of the various structures and components. However, the staff ultimately noted that the seismic assessment of the ESBWR standard plant design does credit only safety-related SSCs and the diesel-driven fire protection pump. This pump is designed to seismic Category II requirements. All SSCs relied upon to address the design-basis seismic event are designed to withstand the effects of the SSE in accordance with the requirements of DCD Tier 2, Section 3.7, which provides reasonable assurance that these SSCs will achieve the seismic margin. The staff found the seismic margin assessment with regard to RTNSS components to be acceptable.

DETERMINISTIC ATWS AND STATION BLACKOUT EVALUATION

The ESBWR is designed to cope with a station blackout for 72 hours. The applicant's analysis demonstrated that reactor water level is maintained above the top of active fuel by operation of the Isolation Condenser System, which is a safety-related system and not a candidate for treatment under RTNSS. The staff found this analysis to be acceptable. (Similar analyses have been used for several older BWRs, which are equipped with isolation condensers.)

The ATWS analysis differs from that of existing BWRs. Under 10 CFR 50.62, boiling-water reactors (BWRs) must have (1) an automatic recirculation pump trip, (2) an ARI system, and (3) an automatically initiated SLC system for ATWS prevention and mitigation. The ESBWR does not use recirculation pumps as in the current BWR fleet, so the recirculation pump trip logic does not exist. Instead, the ESBWR uses natural circulation along with automatic feedwater control. Thus, the ESBWR has implemented an automatic feedwater runback (FWRB) feature under conditions indicative of an ATWS event. This provides a reduction in water level, core flow, and reactor power similar to the recirculation pump trip. This feature is judged to be a major contributor to preventing reactor vessel overpressure and possible short-term fuel damage for ATWS events.

The ESBWR has an ARI system with sensors and logic that are diverse and independent of the RPS. The ARI hydraulically scrams the plant using the three sets of air header dump valves of the CRD system. The ARI logic is implemented in the DPS.

The ESBWR has the required automatic initiation of the SLC system under conditions indicative of an ATWS. The ATWS/SLC system mitigation logic provides a diverse means of emergency shutdown using the SLC for soluble boron injection. The ESBWR design uses electrical insertion of fine motion control rod drives with sensors and logic that are diverse and independent of the RPS. A non-safety system may perform this ATWS diverse automated backup function if the system is of sufficient quality to perform the necessary function(s) under the associated event conditions, as described in the enclosure to Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment That Is Not Safety-Related," dated January 16, 1985. The ATWS mitigating logic system is implemented with the safety-related and non-safety-related DCIS. The non-safety-related DPS processes the non-safety-related portions of the ATWS mitigation logic and is designed to mitigate the effects of potential digital protection system common-cause failures. The DPS transmits the FWRB signal from the ATWS mitigation logic to the feedwater control system. The non-safety-related portions of the ATWS mitigation logic have been identified as requiring regulatory treatment in accordance with the RTNSS process.

The applicant selected the ARI system, the FWRB logic, and the ATWS initiation controls for the SLC system as RTNSS equipment. Based on its review, the staff found this to be acceptable.

ADVERSE SYSTEMS INTERACTIONS

The NRC staff reviewed the description of the evaluation of adverse systems interactions provided by the applicant. In its response to RAI 22.5-17, the applicant described the systematic approach. Passive safety functions are evaluated to identify target areas or components that could be affected by an adverse condition. The systems that interface with each passive safety function are identified to determine if there are non-safety-related SSCs that could potentially cause a failure of a passive safety function. Each interface between a non-safety-related SSC and a passive safety function is evaluated for potential adverse effects. Both functional and spatial interactions are addressed. Spatial interactions are further addressed in the development of the fire and flooding portions of the PRA model. The result of the systematic evaluation is the identification of non-safety-related SSCs that could cause adverse system interactions, and these SSCs would be considered for additional regulatory oversight.

In response to RAI 22.5-17, the applicant stated that the result of the evaluation of the ESBWR did not identify any SSCs that should be considered for its RTNSS program.

In Supplement 1 to RAI 22.5-17, the staff requested that the applicant explain how potential adverse system interactions for non-safety-related components from functional or spatial interactions will be identified and addressed later, during the detailed engineering and construction phase, to ensure that the functions of safety related and RTNSS systems will not be adversely impacted. The staff is currently reviewing GEH's response to RAI 22.5-17 S01. RAI 22.5-17 is being tracked as an open item.

Because of the open item that remains to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

POST-72-HOUR ACTIONS AND EQUIPMENT

The ESBWR is designed so that passive systems are able to perform all safety functions for 72 hours after an initiating event without the need for active systems or operator actions. However, after 72 hours, non-safety-related systems can be used to replenish the passive systems or to perform safety and post-accident recovery functions directly. The following safety functions are relied upon in the period following 72 hours after an accident:

- decay heat removal
- core cooling
- control room habitability
- post-accident monitoring

SSCs needed for actions beyond 72 hours are designated as “Category B.” Within this category, a system, structure, or containment will be designated as B1 or B2, depending on its risk significance. B1 SSCs are necessary to meet the safety goals.

The staff reviewed the augmented design standards described in ESBWR DCD Tier 2, Revision 4, Section 19A.8.3. The staff finds that the design of RTNSS SSCs meeting Criterion B1 in accordance with seismic Category II requirements provides reasonable assurance that these SSCs can perform their function following a seismic event and, therefore, is acceptable.

However, the staff raised several questions regarding the B2 SSCs. The B2 structures are required to meet the International Building Code, which essentially means that they would be able to survive a ground motion equivalent to two thirds of the ground motion due to a 2500-year seismic event, but “survive” is defined as safe to occupy, but some repair would be required before restoration to normal service. The staff’s position was that equipment housed in such structures could be expected to experience more damage.

For example, the electrical building is an RTNSS structure. It houses two non-safety-related standby diesel generators, and it provides space to the Technical Support Center. The electrical building is non-safety-related, non-seismic, and is designed to the Criterion B2 augmented design as described above. The staff does not believe that the B2 level is sufficient to provide reasonable assurance that this equipment will function after an SSE event. This is currently being tracked as an open item.

Similar concerns arose over the possible vulnerability of these structures to wind and hurricane missiles, and over systems designated as “support” which are also in such structures. These also are currently being tracked as open items.

COMMENTS FROM THE JUNE 3, 2008 SUBCOMMITTEE MEETING

The subcommittee meeting reviewed Chapters 19 and 22 of the DCD and the draft SER's. The following points summarize comments from the subcommittee and consultants.

1. As a general comment, the PRA appears to be of adequate quality for the purpose of meeting the commission objectives for a design certification. However, the subcommittee has not reviewed the Level-1 or Severe Accident management details sufficiently to conclude that important elements have not been omitted. In addition, a seismic 'margins' analysis was performed in-lieu of a seismic PRA. (This author would characterize the PRA as giving the lower bound of the risk from the design, admittedly orders of magnitude below the criteria).

2. The ESBWR Regulatory Treatment of Non-Safety Systems seems to comply with the regulations, although without a complete seismic analysis selection was qualitative in some cases. We also need to better understand how such systems are treated in practical terms. For example, if the diesel generators are in this RTNSS category, how are they to be started and loaded before and/or after the 72-hour post-accident phase of the accident?

3. The subcommittee expressed concerns about the potential for adverse interactions between active systems and passive systems, e.g., during the transition from active to passive systems. GEH noted that analysis of such adverse interactions is underway and that system design changes in DCD (Revision 5) were made as a result. The ACRS should hear the details of how this analysis was done and the specific results.

4. One of the most uncertain elements of a passive design PRA is the failure frequency of the various passive safety systems. We do not have a sufficient database for this, at this time. Dr. Apostolakis noted a recent EPRI study on this topic and that NRC had previously attempted a research project to establish bounds on such failure frequencies. Both of these need to be reviewed by subcommittee.

5. The PRA included digital I&C reliability in the risk analysis, by assigning a probabilistic failure frequency to these events. There is no technical justification for values used and this may not be the proper way to characterize software failures, without failure mode insights.

6. The reliability of the squib valves, which are used in a number of circumstances in the design, was discussed. The subcommittee would like to see the database that led to the reliability quantification in the PRA.

7. The subcommittee wants to better understand the fidelity and robustness of the fault tree construction and its quantification. In some sense, the subcommittee would like to do a 'spot-check' audit. The goal is to understand the technical quality, the degree of completeness, and certain details of the PRA, and how the Staff's review has addressed these topics. One example that was brought up in the discussion was GEH model of the GDCS, where the assumption was made not to model the upstream manual valve in the PRA failure analysis.

8. The PRA has used sensitivity analyses as a way to determine the importance of certain systems, associated reliability data. The subcommittee feels that an uncertainty analysis is likely to be warranted now as well as later as the PRA is used for other purposes (ITAACs).

9. There is a key concern about the ESBWR source term. As the committee understands it, MAAP treats the iodine source term as an aerosol that gets permanently removed via deposition processes as well as efficacy of sodium-pentaborate. Based on comments by Dr. Powers' and Dr. Kress' current understanding and the PHEBUS results, it appears that iodine can continuously be re-volatilized into the containment for long periods of time and be available for release to the environment as a gaseous species. The safety implications of this observation needs to be addressed. The staff has promised to give the subcommittee its analysis on this.

10. The staff has used MELCOR to "audit" a number of the Level-2 sequences that were developed by GEH using the MAAP code. Given time constraints during the subcommittee meeting, we were not able review the details of how these results compared. The subcommittee should review these at some later meeting as part of the severe accident management discussions.

11. In past severe accident real-material experiments, it has been observed that control blades melt sooner than the fuel and cladding. This has raised concerns with potential water recovery and reflood events, due to recriticality. Has the GEH analysis verified that sufficient boration is available in all the water sources so that recrticality will not occur? Do MAAP analyses show a different behavior than what is seen in such experiments?

12. There was a concern raised about poisoning of the passive autocatalytic hydrogen recombiners. Has GEH taken into consideration such possibilities for Pd-based systems?

COMMENTS FROM THE AUGUST 21-22, 2008 SUBCOMMITTEE MEETING

The subcommittee meeting reviewed Chapters 19 and 22 of the DCD and the draft SER's. This was a second meeting on this topic (previous meeting was on June 3rd, 2008). Based on the member comments and on the intention to issue an interim letter, some interim conclusions are:

- The ACRS awaits the staff's completed review of the PRA-Rev3 (with open items resolved and errors corrected) to conclude it's adequate for ESBWR Design Certification.

- The ACRS has some issues that need to be clarified to assure the BiMAC functionality as a 'defense-in-depth' measure under transient conditions during the severe accident.

PRA Specific Comments:

1. Overall, the PRA generally appears to be of adequate quality for the purpose of meeting the commission objectives for a design certification. However, the subcommittee has noted certain errors of omission and commission that detract from its quality. These errors need to be corrected and the NRC staff needs to review the current Revision 3 PRA to assure its adequacy.

2. BLEY and STETKAR additional comments and details

Severe Accident Management Comments:

3. DCH is not an issue for this ESBWR design with highly redundant depressurization schemes.

4. Ex-vessel FCI's do not seem to be an issue as a mechanism to harm the BiMAC embedded tubes, but asymmetric melt pours into the water pool after initial melt deposition and deluge actuation could result in energetic FCI's that would 'crimp' the BiMAC downcomer tubes and could affect long-term coolability.

5. The initiation of BiMAC operation and the melt transient deposition could pose problems to the long-term operability of the BiMAC functionality as a 'defense-in-depth' device.
    - First, it is not clear what the sacrificial material composition is to be and how it will be able to handle a high transient heat flux, when melt pours onto a specific localized region (this would be enhanced by a metallic melt pour and/or a large pour rate).
    - Second, the GEH documentation does not seem to indicate any review of past MCCI experiments for transient pouring or melt spreading to bounding this initial heat load.
    - Third, the GEH documentation does not seem to indicate any analysis of an initial asymmetric pour that would inhibit melt spreading and might cause an excessive heat flux focusing effect on the BiMAC vertical/horizontal wall corners.
    - The onset of flow instabilities (static and/or dynamic) may inhibit local cooling and be more limiting than a steady-state CHF limit. Was this possibility considered?

6. The scaling laws for the heat transfer experiments conducted were not specifically discussed.

EXPECTED COMMITTEE ACTION

The Committee will review this matter and will issue an interim letter to the EDO on Chapters 19 and 22.

References

1. SER with open items for Chapter 19, "Probabilistic Risk Assessment and Severe Accidents," ML080850473

2. SER with open items for Chapter 22, "Regulatory Treatment of Non-Safety Systems," ML080940707

3. NEDO-33210, "ESBWR Certification Probabilistic Risk Assessment," Rev. 3, May 2008

4. T. G. Theofanous, "The MAC Experiments: Fine Tuning of the BiMAC Design," NEDE-33392P, March 2008

Attachment

June 6, 2008

Consultant's Report on the June 3, 2008
ESBWR Subcommittee Meeting

T. S. Kress

BACKGROUND

The purpose of this meeting was for the ESBWR Subcommittee to review Chapters 19 and 22 of the staff's Safety Evaluation Report. Chapter 19 deals with the ESBWR "Probabilistic Risk Assessment and Severe Accidents" and Chapter 22 deals with "Regulatory Treatment of Non-Safety Systems."

This author has a conflict of interest with respect to the BiMAC system and will refrain from making any comments on this system.

COMMENTS

1) As a general comment, the PRA appears to be robust and of acceptable quality. However, the Subcommittee has not reviewed the details of the fault trees sufficiently to conclude that important elements have not been omitted. In addition, Seismic events were dealt with via the "seismic margins" process. Consequently, the PRA must be considered as being incomplete with respect to the risk determinations and the determinations of components that might qualify for RTNSS.
2) There did appear to be appropriate use of the PRA to meet the Commission's "objectives" for a PRA for new certifications and it appears that these objectives have been met for the ESBWR.
3) The PRA attempted to include digital I&C reliability in the risk determinations by assigning a probabilistic failure frequency to these. There is no technical justification for such a process.
4) It was gratifying to hear that the staff had used MELCOR to "audit" a number of the Level-2 sequences that were developed by GEH using the MAAP code. We were not shown the details of how these results compared. The Subcommittee should review these at some later meeting.
5) It was interesting that a LRF of 10^{-6}/yr is now considered to be a Commission "goal". I recall that the staff previously rejected this concept because of the inability to properly define an LRF and because most definitions resulting in LRF superceding the QHO Safety Goals. We were told that a CDF accompanied by a containment failure was considered to be a LRF. Given the unresolved issue of continuous pumping of fission product iodine into a leaking containment (see Comment -10 below), this may be an insufficient definition. This is particularly so because I personally view an LRF as a potential surrogate for societal risk and have tentatively concluded that a value of about 10^{-6}/yr would be a potential acceptance criterion on par with the QHO Safety Goals.
6) There was considerable discussion at the Subcommittee meeting about what should be done about a hypothetical possibility that the PRA used and approved for certification might not include something that is later (say at the COL stage) decided to be important enough that it is deemed that it should have been included as part of the certification. My view on

this is that once a design is certified, to change the details of that will have to be done via the backfit process. Any possibility like the above should be dealt with at the COL stage using the COL PRA.

7) The Subcommittee expressed concerns about the potential for adverse interactions between active systems and passive systems especially during the transition from the former to the latter. It was noted by GEH that a search was made for such adverse interactions and that system design changes were made as a result. The ACRS should hear the details of how this search was done and the results.
8) One of the most uncertain elements of a passive design PRA is the failure frequencies of the passive systems. We do not have a sufficient data base for this. We were told that EPRI had made a study of this and that NRC had previously attempted a research project to establish bounds on such failure frequencies. Both of these need review by ACRS.
9) In an earlier meeting, I had raised an issue about the potential for long term recriticality because of the boiloff of boric acid that had been injected into the core region for shutdown. I now know this is not an ESBWR issue because ESBWR injects sodium pentaborate instead of boric acid. This, however, remains as a possible generic issue for PWRs which do use boric acid for this purpose.
10) The most overriding concern about the ESBWR source term is the MAAP treatment of fission-product iodine as an aerosol that gets permanently removed via deposition processes. Based on current understanding and the PHEBUS results, it appears that iodine can continuously be re-volatilized into the containment for long periods of time and be available for release to the environment as a gaseous species. The safety implications of this needs to be addressed. The staff has promised to give the Subcommittee its views on this.

Attachment

June 24, 2008

Consultant's Report on the ESBWR Subcommittee Meeting on DCD Chapter 3: "Design of Structures, Components, Equipment, and Systems

T. S. Kress

The staff is very good at reviewing the compliance with regulations related to this chapter of the DCD and they have done a thorough and competent job with their review. I do not believe that the remaining open items are of sufficient import to significantly impact the approval of this chapter.

I have only one potential issue with the seismic analyses related to elevated pools. We were told that, to account for sloshing effects, a fixed fraction of the water is assumed to "slosh" and the remaining water treated as a solid mass moving with the tank at the various seismic frequencies. Intuitively, one would not expect the quantity associated with sloshing to be a fixed fraction. The amount should vary with the aspect ratio of the tank as well as to other geometrical factors (such as the tank not being a right circular cylinder) and the applied frequency. Where one would expect this to have an effect is on the resonant frequency of the tank/water system as well as the associated forces on the support structures. I recommend that ACRS further review this issue.