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

July 13, 2000

MEMORANDUM TO: ACRS Members

FROM: P. Boehnert, Senior Staff Engineer PB

PB FOR T. Kress, Chairman, Severe Accident Management Subcommittee

PB FOR R. Seale, Subcommittee Member

SUBJECT: COOPERATIVE SEVERE ACCIDENT RESEARCH PROGRAM MEETING (CSARP 2000), MAY 8-11, 2000 - RESIDENCE INN HOTEL, BETHESDA, MARYLAND

The Year 2000 CSARP meeting was held on May 8-11, 2000 in Bethesda Maryland. T. Kress and R. Seale attended the May 8-10 Sessions. P. Boehnert attended portions of the same sessions[1]. Below is a combined report of our impressions of the meeting presentations.

Plenary Session 1 - Panel Discussion on Current Understanding of Severe Accident Phenomena and its Adequacy from a Risk-Informed Perspective

A. Thadani moderated the subject Panel, consisting of F. Eltawila, RES, J. Peltier, IPSN, France, B.R. Sehgal, RIT, Sweden, T. Theofanous, UCSB, and, M. Vidard, Edf, France. Basically, the Panelists provided response to a set of questions posed by the NRC regarding the adequacy of current understanding of key severe accident phenomena and the need for additional research. Key points noted included:

- All agreed that successful ex-vessel debris coolability is questionable, given the inconclusive experimental results to date. Theofanous said that the industry needs to think "outside the box" here, specifically with regard to use of new (high

[1] Our attendance was restricted by the schedule conflict with the May 11-13 ACRS Meeting.

temperature) materials[2]. The efficacy of molten debris retention in the reactor vessel lower head is also questionable for operating plants, as flooding of the reactor cavity may not always be possible, or if possible may not be successful in preventing vessel failure.

- DCH and hydrogen generation and combustion are generally considered "resolved" with respect to plant risk.

- Fission product release and transport was considered to be fairly well in hand, but concern was expressed regarding iodine chemistry (see below).

- In response to the question: "Can uncertainties be dealt with in a bounding manner using currently available tools?", there was no consensus (some "no's", some "yes's", some "maybes's").

- In summary, it was agreed that few issues remain to be addressed for severe accidents. Those that do can be addressed in a integrated manner, without additional large-scale experiments. It was noted, however, that no program exists to address ex-vessel debris coolability, which remains a significant open issue. Dr. Kress also expressed concern regarding the ex-vessel debris coolability issue.[3]

Technical Sessions

The bulk of the meeting was spent in review of the various on-going phenomenological research programs of the members countries. Highlights of the presentations I attended included:

- The OECD is conducting additional testing at Sandia on vessel lower head failure phenomenon. The objectives of these tests are to characterize the mode, timing, and size of RPV lower head failure for conditions of low RCS pressure with large delta-T's across the lower head wall and for pressure transients, all without any external cooling. Five experiments are planned (5 Mpa & 2 Mpa - both uniformly heated, effects of pressure, penetration, and vessel erosion). The first test was scheduled for June 2000.

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[2] Theofanous also indicated that issues pertaining to failure of steam generator tubing under severe accident conditions needs careful evaluation, documentation and review, (he was describing the DPO issues that have been raised by Dr. Hoppenfeld in RES).

[3] In a sidebar, Dr. Kress expressed concern over air ingestion accidents and the impact on the source term.

- The current status of the OECD RASPLAV program was noted. The major activities of the program were experimental studies of prototypic molten core materials in a vessel lower head, use of simulate fluids (e.g. salt) in small-scale tests to support analysis of the large-scale (200 kg) integral melt tests, and code development (CONV3D code). All the testing has been completed. Main test results noted included: (1) establishment of a convective molten pool at a temperature level of 2600 - 2650 °C, (2) corium stratification was observed in two of the four large-scale tests, with zirconium enrichment seen in the upper layer (3) establishment of a homogeneous molten pool in the two other large-scale tests and (4) separation of zirconium oxide in the last large-scale test (via relocation of molten fractions through gaps and cracks). Items remaining to be concluded for this program are the analysis and interpretation of the test data and report preparation.

- An overview of the European Commission (part of the European Union organizational structure) severe accident projects was discussed. A total of 16 projects are underway in the areas of corium formation and management, reactor pressure vessel behavior to corium impact, source term issues, hydrogen behavior and management in containment, containment by-pass sequences (including tests to evaluate accident management intervention for steam generator tube rupture scenarios) and code development. These programs began earlier this year and have an average duration of three years. Total costs is ~ \$25M, with the EU contributing ~ \$12.5M. Nine of the above programs are related to the Phebus fission product (FP) test program (see below).

- The status and results to date of the Phebus-FP Program were noted. Three tests have been run to date - the last (FPT-4) conducted on July 22, 1999. Three more tests are planned in the current program, the last to be run in the 2004-05 time frame, and there is talk of a follow-on program as well ("Phebus-2000"). The attached figure (Figure 1) gives details on the test matrix and the facility. Some of the observations cited from the tests performed included:

  ○ maximum hydrogen production rate is strongly coupled to the thermal behavior of the cladding degradation process

  ○ ruthenium was released from the fuel but deposited out just above it (low volatility)

  ○ re-emission of deposited fission products may occur during the late stage of an accident if high steam flow rates are introduced

  ○ results for iodine behavior suggest the need for new experimental data (separate effects tests) and code models, as evidence of non-equilibrium mechanisms was seen. Other points noted were that iodine becomes rapidly insoluble in the sump water due to reaction with oxidized soluble silver and/or silver colloids. Production of volatile iodine by radiolytic oxidation reactions was characterized as very low. For the short term, the gaseous iodine production depends mainly on the break injection. The gaseous iodine reacts with paints and is partially trapped and partially converted into released organic iodides. Organic iodides are the main contributors to the middle source term (few days post accident). The effect of boron carbide on the iodine will be evaluated in test FPT-3 scheduled for 2003.

○ regarding the latest test run (FPT-4), it was designed to be as study of: (1) low volatile fission products, actinides, and uranium releases from a debris bed; (2) high volatile fission product releases from a debris bed - contrasting with a previous experiment using a rod bundle; (3) the physics of a debris bed, with a focus on heat transfer mechanisms; (4) the transition of a debris bed to a molten pool, and (5) fission product releases from a molten pool[4]. The test conduct was successful in all significant respects. Very little in the way of results was provided. It was noted that a small amount of volatilized fuel was trapped in the filters (few percent). The pool melt progression was fast. Non-destructive examinations have been performed and post-test analyses are underway. It is planned to use codes to scale-up the test results to full-size plants.

- Two experimental programs being conducted at JAERI (Japan Atomic Energy Research Institute) were discussed: the ALPHA (Assessment of Loads and Performance of a containment in a Hypothetical Accident) and VEGA (Verification Experiments of radionuclides Gas/Aerosol release) Programs. For the ALPHA program, experiments are in progress to study melt fragmentation, melt jet breakup, and melt entrainment. Steam explosion experiments using molten silicon[5] are under design. Two codes are being developed: CAMP and JASMINE for assessment of ex-vessel fuel-coolant interactions. The VEGA work is focused on study of the mechanism of fission product release from PWR and BWR fuel, including MOX, and to improve the predictability of the source term. Two VEGA tests have been run; results were shown from the first test (Figure 2), and compared to similar tests run at Oak Ridge under the management of Dr. Kress.

  Dr. Kress complemented JAERI on the VEGA test program. There was discussion regarding the significant discrepancy between the ORNL and VEGA tests on the

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[4] Dr. Kress in a "sidebar" to me, noted that he didn't expect any substantial FP releases will be seen from the molten pool.

[5] Dr. Kress notes that sufficient theoretical basis exists to interpret simulant results in terms of prototypic materials.

release fractions of two isotopes (ruthenium - Ru-106, and praseodymium - Pr-144). The JAERI representative said that they suspect a measurement error on their part.

- A Japanese organization affiliated with Toshiba performed an interesting set of experiments to evaluate failure criteria and fission product "trapping effect" for containment penetrations under severe accident conditions. Specifically, tests were performed on prototypical containment penetrations used on the ABWR, including the gasket seal for the equipment hatch and main flange, and low-and high-voltage electrical penetrations. "Severe accident conditions" (SAC) was defined as 200° C and twice design pressure. The test facility and test procedure are shown on Figure 3. Conclusions from the tests were:

  ○ All prototypical penetrations were intact under SAC conditions
  ○ Penetration failure was governed by temperature, not pressure
  ○ The margin-to-failure temperature was 70° C above SAC conditions
  ○ The decontamination factor for fission product trapping was estimated to be 380
  ○ Total leakage area from the tests for failed components ranged from 1/7 - 1/34 times smaller than that assumed in current severe accident analysis codes

<u>Remarks by Commissioner Diaz</u>

Commissioner Diaz made some extemporaneous remarks at the opening of the penultimate meeting session on May 10. Key aspects of his remarks were:

○ Cooperative international programs, such as CSARP, have become a necessity given resource constraints and the small size of the nuclear community

○ The CSARP work is helping elevate technical capabilities around the world. There is a need to address the uncertainties associated with severe accident phenomena. The nuclear industry will be well served if the relevant SA phenomena are bounded.

○ The criticality accident at Tokaimura was not "bounded" in the sense of what the public was told by the media, versus the facts of the situation; i.e., criticality events are localized. Likewise, the SA research needs to bound the outcome of such an accident.

○ The NRC has entered a new regulatory regime -- a risk-informed approach. Overseas, the common perception is that the NRC is advocating a risked-based regulatory regime. This is not the case. The results of the CSARP programs will aid the NRC in striking the proper balance among PRA, defense-in-depth and deterministic regulation.

In a "Q & A" session, Dr. Diaz said that he supports continuation of SA research, to the extent it meshes with NRC's overall regulatory scheme. Citing the work that lead to the revised source term rule, he said that the challenge of the CSARP is to keep its work relevant to the "big picture". Dr. Theofanous asked if Dr. Diaz felt that work on ex-vessel coolability was important. Diaz responded that he believes it is very important and that he would pursue this matter with the staff. Dr. Kress asked that, in light of the progress made in SA research in recent years, whether Dr. Diaz believes that there is a need for the Commission to issue a new policy statement on risk-informed regulation. Dr. Diaz agreed that this may be the case and said he'd discuss this matter with the Commission.

In closing, Dr. Diaz said that past events drove the Agency to construct a patchwork of regulations. Recent events showed that change was imperative. The current move to risk-informing the regulations is not a relaxation of same, instead, resources will be directed to the proper areas of safety.

<u>Panel Discussion - Current Capabilities of Severe Accident Codes for Plant Applications</u>

The Panelists for this discussion were: R. Henery, FAI, J. Micaelli, France, P. Schmuck, Germany, C. Tinkler, NRC, and G. Holahan NRC - Moderator. The panelists provided responses to a set of questions as follows:

- Do the Codes Provide Adequate Details and Sufficient Accuracy for Plant Applications?

  Mr. Schmuck indicated that a lot depends on the definition of the terms "adequate details" and "sufficient accuracy". For the latter, he indicated that the lack of data for the later phase of a SA means that accuracy can only be estimated "roughly"

  Mr. Tinkler said that system-level and control volume codes are adequate, provide they are used within the applicability of the test data. Both Mr. Tinkler and Mr. Micaelli noted that code validation efforts are flagging due to the costs involved and the lack of large-scale test data.

  Dr. Henry said that the first rule for code use should be "do no harm"; i.e., the codes should not mislead.

- Are the Physical Models Consistent With Current Understanding of Phenomena?

  The answer was "generally yes" but was qualified by noting that there are incomplete or missing models in some codes. Dr. Henry said that he is more comfortable with modeling on a global (or integral) basis, particularly as more comparisons to experiments are made. Likewise, Mr. Micaelli said that codes cannot "stand alone"; they must be tied to engineering evaluations of test data and other studies.

Mr. Tinkler said that codes can analyze a wide range of moderate or severe accidents in an integrated manner, but uncertainties remain. These uncertainties can be accommodated by user control of parameters (tuning?), provide that <u>users are knowledgeable of severe accident phenomena</u> (emphasis added).

- How are Phenomenological Uncertainties Dealt With in the Codes? Is the Treatment of Uncertainties Such That Some Bounding Analyses Can be Performed Using the Codes?

  Mr. Micaelli said that model uncertainties can be dealt with by coupling the codes with statistical tools. Phenomenological uncertainties must be dealt with by use of a bounding approach. Dr. Henry recommended use of a best-estimate approach with boundaries on the uncertainties. Mr. Tinkler recommended use of bounding calculations; he said that uncertainties can be quantified with effort and when appropriate via consideration of the relevant driving phenomena.

- What is the Validation Status of SA Codes? What is the Nature of the Assessment Data Base?

  Dr. Henry said that much more remains to be done here. He urged continuous comparison with the TMI-2 data as this is one of the best "data points" we have. The international standard problems are also crucial. Validation is a dynamic process and code comparisons with data should be repeated on a periodic basis.

  Mr. Micaelli said that much more work is needed here, citing the situation with modeling of iodine chemistry ("poor").

- How Extensively Have Severe Accident Codes Been Used for Plant Safety Analyses and Plant Risk Assessments? How Have the Codes Been Used for Regulatory Decision Making?

  Mr. Schmuck said that Germany has applied SA codes to the Konvoi and European PWR plant designs. Mr. Micaelli said the France also applied these codes to the EPWR design as well as evaluation of hydrogen recombiner designs in its PWRs.

- What Are Your Views on Further Development and Maintenance of Severe Accident Codes?

Mr. Schmuck advocated development of a new comprehensive integral SA code, building on MELCORE, which he called S(Super)-MELCORE. He suggested that the code be modular to allow modeling contributions from the international community, and use a popular and efficient programming language.

Mr. Micaelli said that the code budget is being cut back in France, so emphasis has switched from development of new codes to maintaining current codes and ensuring the continuity of the work teams. For the long term, work will focus on: treatment of stochastic processes, direct numerical simulation, and connection of current codes with other "domains" (e.g., CFD codes)

Mr. Tinkler said that NRC's focus is on use of integral system codes.

Dr. D. Powers also provided his comments on the above questions[6] [7]. A copy of his write up is attached.

MELCORE Code Development and Assessment

- A Sandia Laboratory representative provided an overview of the MELCORE 1.8.5 version code release, scheduled for May 30, 2000. New models/improvements to MELCORE include: addition of an iodine aqueous chemistry model (Dr. Powers was involved in this work), a passive autocatalytic recombiner model, improved hygroscopic aerosol model, improved initiation sequence for bottom head model, and improved features for the PLOT model. A MELCORE/CONTAIN parity assessment was recently completed. Overall, MELCORE's containment models were characterized as performing very similarly to those of CONTAIN. Code improvements noted for subsequent versions of MELCORE include incorporation of reflood modeling capability, radial melt relocation modeling, and melt/crust modeling improvement in the core and lower head regions.

- Details of the improvements in core degradation modeling capability for MELCORE 1.8.5 were discussed. Improvements detailed included addition of multiple options

[6] One question in the list ("Is There a Common Understanding Between the Code Developers and Code Users as to a Desired Level of Accuracy for Plant Safety Applications and Regulatory Decision Making") was not discussed at the meeting. Dr. Powers answered this question as "No".

[7] Dr. Kress said that he is in complete agreement with the views expressed by Dr. Powers.

pertaining to modeling of core structures, new core structure support model (for P and BWRs), particulate debris exclusion model (to limit particulate debris entering fuel bundles), improved BWR flow blockage model, and an upgraded candling model. The consequences of these model upgrades are: BWR fuel remains supported longer, fission product release is increased, and hydrogen production by oxidation is increased as well.

I have copies of all the slides/handouts from the meeting; please let me know if you'd like a set.

Attachments: As Stated

cc: Balance of ACRS Members
R. Savio

cc w/o attach (via E-mail):
J. Larkins
H. Larson
S. Duraiswamy
ACRS Technical Staff & Fellows

# THE PHEBUS FP TEST MATRIX

| No. | Type of fuel | Fuel bundle | Primary circuit | Containment vessel | Date |
|---|---|---|---|---|---|
| FPT-0 | Fresh fuel, 1 AgInCd rod, 9 d. pre-irrad. | Melt progression & FP release in steam rich environment | FP chemistry and deposits in non condensing steam generator. | Aerosol deposition. Iodine radiochemistry at pH 5. | Dec. 2, 1993 |
| FPT-1 | BR3 fuel ~23GWd/tU, 1 AgInCd rod, re-irrad. | As FPT-0 with irradiated fuel. | As FPT-0 | As FPT-0. | July 26,1996 |
| FPT-2 | As FPT-1 | As FPT-1 under steam poor conditions. | As FPT-1 with effect of boric acid. | $H_2$ recombiner, pH9, evaporating sump, | 2000 |
| FPT-3 | As FPT-1, with B4C instead of AgInCd | As FPT-2 | As FPT-0 | As FPT-2? | 2003 |
| FPT-4 | EdF fuel ~33GWd/tU No re-irrad. | Low volatile FP & actinide release from $UO_2$ $ZrO_2$ debris bed, up to melting. | Integral filters in test device Post-test studies on samples | | July 22,1999 |
| FPT-5 | As FPT-1 | Fuel degradation and FP release in air conditions. | Deposition & chemistry of FPs in air conditions. | As FPT-1 or 2 | 2004-2005 |

# THE PHEBUS-FP FACILITY

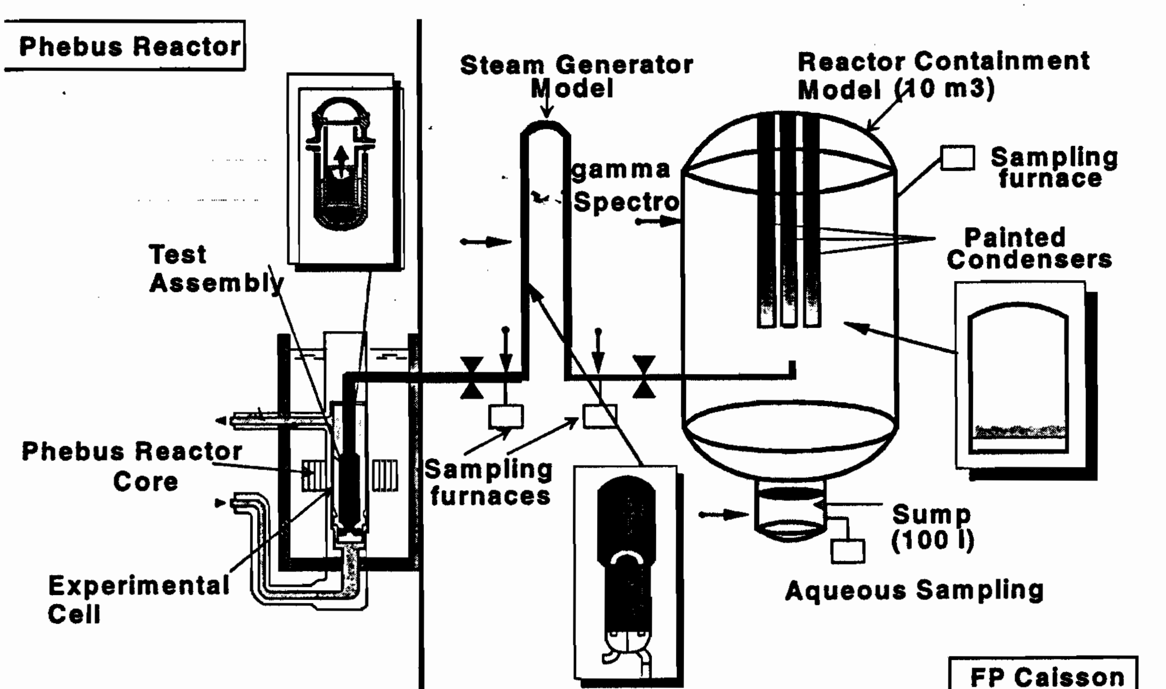


FIGURE 1

## VEGA-1 Total Releases Estimated from Intensity Changes of FP Gamma Peaks (Assuming no release of Eu-154)

Antimony

| Nuclide | γ Energy | Abundance | Before | After | Released | VI-3* |
|---|---|---|---|---|---|---|
| | keV | % | counts/$10^3$s | counts/$10^3$s | % | % |
| Sb-125 | 429 | 29.4 | 110 | 10.5 | 96 | 99.2 |
| Cs-134 | 605 | 97.6 | 23752 | 3278 | 99 | 99.9 |
| | 796 | 85.4 | 18130 | 5522 | 91 | |
| Cs-137 | 662 | 85.2 | 71827 | 14374 | 87 | |
| Rh-106 | 622 | 9.79 | 774 | 384 | 70 | 5 |
| (Ru-106) | 1051 | 1.6 | 77 | 72.3 | 65 | |
| Pr-144 | 696 | 1.34 | 44 | 26 | 61 | <0.2 |
| (Ce-144) | 2186 | 0.7 | 16 | - | - | |
| Ag-110m | 885 | 72.8 | 9 | 5.2 | 71 | 84(VI-1) |
| Eu-154 | 724 | 19.7 | 762 | 1177 | 0 | <0.01 |
| | 1275 | 35.5 | 1105 | 4242 | 0 | |

1 order meas. error

ok?

* ORNL Test VI-3: 2427°C(20min) in steam

Marker for fuel release

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# Summary of VEGA-1 Test

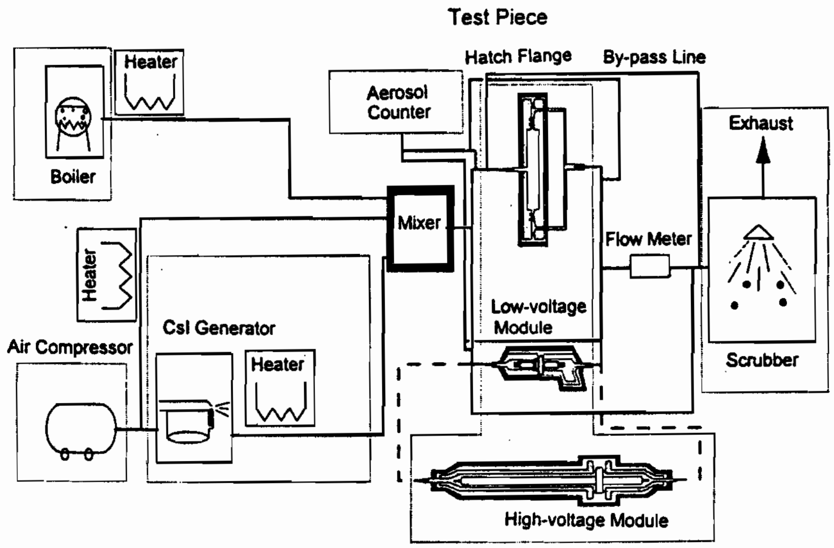
- First heating test VEGA-1 was conducted on Sep. 9 '99.
- Two PWR fuel pellets at 47GWd/tU were heated up to 2500°C for 10min in an inert atmosphere at 0.1MPa.
- Gamma detectors watching at the fuel and the filters gave the release histories of radioactive fission products, e.g. Cs-137, Cs-134.
- Total releases of Cs-134, Cs-137, Sb-125, Ru-106, Ag-110m, Ce-144 were preliminary estimated from spectrum change of the fuel.
- Off-line chemical analyses of the apparatus will give information on release of fission products with better accuracy.
- Gamma measurement of charcoal trap for Kr-85 was not performed properly due to problems in collimator shape. Qualitative concentration trend in the loop was measured.

FIG. 2

# Features of Test Facility

143 psi

- Test pieces can be raised up to 400℃ and 1MPa to simulate SA conditions
- CsI aerosol was generated using an atomizing type of aerosol generator
- CsI aerosol concentration and its diameter were measured using an optical particle counter at inlet and outlet test piece
- Leakage flow rate was measured by a mass flow meter




Nupec

# Test Procedure

**(1) Integrity Confirmation Test**

**AM condition**

Confirmation of no change of outlet gas temperature

**(2) Failure Criteria Test**

**High temperature beyond AM condition**

Measurement of change of outlet gas temperature

Evaluation of Leak path area

Measurement of leak flow rate v.s. pressure

**(3) Aerosol Trapping Test**

**CsI aerosol supply into damaged test piece**

Measurement of CsI concentration at inlet and outlet test piece

Evaluation of decontamination factor (DF)

## Test Condition

Temperature : Max. 200℃
Pressure : Max. 0.8 MPa
Radiation Aging : 800 kGy, 0 kGy
Atmosphere : Steam, Air

Temperature : 200 - 400℃
Pressure : 0.4 – 1.0 MPa
Radiation Aging : 800 kGy, 0 kGy
Atmosphere : Steam, Air

Temperature : R.T. - 200℃
Pressure : 0.1 - 1.0 MPa (Max.)
Atmosphere : Air

Temperature : 100℃ - 200℃ (Max.)
Pressure : 0.11 - 0.6 MPa (Max.)
Atmosphere : Air
CsI concentration : 10 - 300 mg/m$^3$
CsI diameter : 1- 2 micron (AMMD)

FIG. 3

Nupec

**From:** "Powers, Dana A" <dapower@sandia.gov>
**To:** "'KRESS, T.S.'" <TSKress@aol.com>, "'SEALE, Robert...
**Date:** Tue, May 2, 2000 6:43 PM
**Subject:** CSARP Questions and Answers

Tom, Bob,

I understand you guys are attending the CSARP. They are planning a panel discussion on codes. Attached are some of my thoughts on the questions they raised.

Dana

<<CSARP Questions and Answers.wpd>>

**CC:** "'pab2@nrc.gov'" <pab2@nrc.gov>

To: Distribution May 2, 2000

From: D.A. Powers

Subject: Questions for CSARP Panel Discussion

Apparently, there will be a panel discussion at the CSARP meeting dealing with "Current Capabilities of Severe Accident Codes for Plant Applications". Some of my thoughts on the suggested questions are provided below:

**1. Is the current generation of severe accident codes capable of simulating the following severe accident phenomena in adequate detail and with sufficient accuracy for plant applications?**

- **core melt progression**
- **melt/vessel, melt/water, and melt/structure interactions**
- **in-vessel and ex-vessel core debris coolability**
- **direct containment heating**
- **hydrogen generation and combustion**
- **fission product release and transport**

**Response:**

**General:** In an era of risk-informed regulation, one would think that the NRC and the licensees would want to know about plant risk in some detail. That is, risk and the changes in risk with plant changes would be of great interest. But, in an era of core damage frequency informed regulation it is not apparent that a very detailed understanding of severe accidents is needed. It may be adequate to know that severe accidents are to be avoided. The current generation of plants have some capabilities to mitigate the consequences of accidents, but since these capabilities have not been designed to deal with severe accidents they are less than perfect. If this is the level of detail that is needed for severe accident analysis codes, what we have is more than adequate.

On the other hand, we have seen instances in the recent past - such as the analysis of steam generator tube integrity issues - where severe accident analyses were of use in deciding whether or not to revise regulations. On these occasions we have seen the codes used to produce results that are difficult to believe. For example, the severe accident codes are used to show that pressurized water reactors will not stay pressurized during core damage progression because of heat transport to the reactor coolant system structures leads to creep rupture of the system and depressurization. But, no such threat to the structures of the TMI-2 reactor coolant system have ever been mentioned. It appears that based on a Bayesian analysis of the one data point that we have, the codes are analyzing the wrong accidents. That is, stylized accidents have been selected that do not recognize the significant features of real accidents. It is not evident to me that the codes can do a good job of analyzing more realistic accidents. Certainly, it seems that they have had to struggle to predict well the phenomena observed during the accident at TMI.

**Core Melt Progression**:

I am not persuaded that the existing models of core melt progression are sufficiently developed for a real understanding of the cores actually in plants. I see two flaws in the existing codes:

a. core damage progression is based on "candling" type processes seen in tests with unirradiated and trace irradiated fuels and do not model explicitly the foaming and swelling of fuel observed in tests with irradiated fuels. The recent PHEBUS FP-4 test may make it clearer how important this foaming phenomenon is since investigators are attributing, now, some of the unexpected data from the test as the result of fuel foaming.

b. the codes still use crude representations of the phase relationships in the pertinent Zr-U-O system to decide when fuel relocates from the core region. Rather unusual gyrations of code parameters seem to be needed each time a test is done that involves fuel relocation.

**Melt Vessel Interactions**

It is not at all clear to me that we have a good understanding of the core debris vessel interactions. Some want to treat this interaction as a natural convection problem, but I cannot imagine how natural convection patterns will develop if core degradation is progressing non-uniformly and materials are continually falling into the pool of core debris interacting with the vessel. It is not clear to me that models of the heat transfer during core debris interactions with the vessel have been done properly since I see no evidence that the models take into account the heats of dilution that accompany ablation of metal from the vessel.

**Melt Water Interactions**

This is a multifaceted issue. The phenomenon of steam explosions springs immediately to mind, and it is true that we still cannot predict when steam explosions will occur nor what the thermal to mechanical energy conversion will be. On a more benign topic, it is not clear to me that the codes are capable of predicting well the phenomena that occur should water be readmitted to a degrading core. I just don't think we have a data base sufficient to tell whether the codes are good enough for predicting what will happen when coolant is readmitted.

**Melt/Structure Interactions**

Without good modeling of heats of dilution and intermetallic reactions, I am not convinced that the codes can model well the core debris structure interactions.

**Debris Coolability**

Once core debris is in a fragmented state, I think the models can predict coolability provided that they recognize that debris beds will have a distribution of particle sizes and that the particles making up a bed are likely to be size stratified. Ex-vessel coolability is a less crucial issue to my mind since the presence of water is very likely to limit the releases of radioactivity whether or not the core debris is coolable. With respect to in-vessel coolability, I am less confident in the codes for situations in which coolant is readmitted to the core before the significant loss of core geometry. I am not persuaded that we have predictive models for this condition.

**Direct Containment Heating**

Because of possible conflicts of interest, I would prefer to not comment on this issue beyond noting that there do seem to be some differences of opinion within the technical community. Also, I refer to my comments above about the likelihood that reactor coolant systems will depressurize prior to reactor vessel failure.

**Hydrogen Generation and Combustion**

All that I have seen suggests to me that the codes do a useful job in the prediction of hydrogen generation when they recognize that steam will react with stainless steel to produce hydrogen as well as reacting with zirconium. I think that we have an adequate understanding of the combustion and detonation regimes of the relevant hydrogen gas systems. I think the issues of hydrogen combustion remain:

- hydrogen distribution under low driving force conditions (while the RCS is pressurized)
- transition from deflagration to detonation

**Fission Product Release and Transport**

To respond to this question, I think we have to distinguish between the systems level severe accident analysis codes like MELCOR and the more specialized codes for predicting fission product behavior like VICTORIA and SOPHAEROS. I will discuss what I think the situation to be for the systems level models. Points that I make are:

- it appears the CORSOR/Booth model may have enough of the salient physical and chemical models to adequately predict the releases of volatile fission products during the earlier stages of core degradation prior to the loss of core geometry. There may be residual questions about the effects of high pressure on the releases. PHEBUS results suggesting higher than expected mobilities for Mo and Ru still need to be addressed and explained. Details of the timing of the releases of volatile fission products could be improved if the release models took into account both the mass transport effects of fuel relocation and the effects of gas composition.
- it is an open question whether we need a model of fission product release from fuel configured in a debris bed similar to that formed at TMI. This question may be resolved once we have the results of the PHEBUS-FP-4 test.

- I don't believe it important to have a model of release from molten pools of core debris within the reactor coolant system.

- Our accident analysis codes predict vessel failure in many cases when only a fraction of the core has significantly degraded. The systems level codes seem not to address what happens to what may be half the core of fuel and the fission product releases from this fuel following vessel failure. Perhaps the need to better understand the post vessel failure release of fission products will become clearer once the results of the PHEBUS FP5 test are available.

- Models of the release of nonradioactive materials during core degradation are pathetic. System level codes don't even have useful models of the releases of Ag, In and Cd from precious metal control rods. There has been no attention given to the vaporization from steel surfaces exposed to high pressure, high temperature steam. Over and over we say that the releases of these nonradioactive materials have important effects on the behavior of fission products, but we never seem to do anything about it.

- I see no evidence that the systems level codes adequately model the late revaporization source term from radionuclides deposited on surfaces within the reactor coolant system. It doesn't appear that the codes even attempt to realistically model the surface speciation of these deposited materials or calculate their vapor pressures.

- The aerosol models used to predict much of the aerosol transport within the reactor coolant system as well as aerosol behavior in the reactor containment are impressively sophisticated. There have been modern developments in modeling simultaneous agglomeration processes and deposition processes that are not yet incorporated in the codes. But, the codes have only been validated by comparison to tests done with nonradioactive aerosols. There is strong evidence that especially in the containment the aerosols produced in reactor accidents will be charged unlike the aerosols used in the tests. Aerosols that do not have a Boltzmann-like charge distribution will be subject to Coulombic forces that are orders of magnitude stronger than the forces that are modeled now in the codes. The ramifications of these forces have not been studied in any meaningful way even by analysis. The one thing that makes aerosols from reactor accidents different than aerosols of more conventional type is not modeled in the accident analysis codes even to the extent necessary to say whether this peculiarity of the aerosols is important or not.

- Our codes have a poor treatment of the attenuation of aerosol releases during bypass accidents such as steam generator tube rupture accidents. Perhaps the tests now being planned in Switzerland will make it more apparent how important it is to model this attenuation in the secondary side of steam generators.

**2. Do the codes embody physical models that are consistent with the current understanding of relevant phenomena?**

The systems level codes probably lag our current understanding of relevant phenomena. Examples are cited in the responses to the first question.

**3. How are phenomenological uncertainties dealt with in the codes? Is the treatment of uncertainties such that some bounding analyses can be performed using the codes?**

I think the model developers have done a rather good job making it possible to adjust models in the codes to address uncertain issues. I think the codes have not been developed to facilitate systematic uncertainty analyses. I would hope a future generation of severe accident analysis codes would make it easier to do such uncertainty analyses.

I am not a fan of bounding analyses. They have their place, but I think they are used a little too often. The problem is that we may not know enough to assure that a bound has really been devised. A particularly obnoxious example is what has been done with respect to the analysis of the bypass accidents. We don't know what attenuation of aerosol releases will occur once aerosols and vapors escape the reactor coolant system in a bypass accident, so we assume that there is no attenuation. The bypass accidents, which have middle of the road probabilities, become risk dominant. Who knows what distortion of safety attentions has been caused by this bounding analysis.

**4. Is there a common understanding between the code developers and code users as to a desired level of accuracy for plant safety applications and regulatory decision making?**

No.

**5. What is the validation status of severe accident codes? What is the nature of assessment data base?**

I think the validation status of the codes is flagging. That is, I think the code developers have really tried to use older data to validate their modeling. But, the base of data for code validation is growing and the code developers are not keeping up. For instance, I have not seen comparisons of the MELCOR predictions to the results of the PHEBUS tests, or to the KAEVER tests, or to the JRC and German tests of core debris-water interaction tests, or to the STORM tests. MELCOR has been compared with good results to some of the RTF tests of iodine partitioning but this work seems to have gone into hiatus as more demanding test results come available for comparison.

**6. How extensively have the severe accident codes been used for plant safety analyses and plant risk assessment? How have the codes been used for regulatory decision making?**

Severe accident analysis codes have BEEN crucial to the development of the NRC Revised Accident Source Term (NUREG-1465). Severe accident analysis codes were crucial to the conduct of the NUREG-1150 assessment of accident risks at five representative plants and the general conclusion that current regulations are consistent with the NRC's safety goal. These are important accomplishments. Severe accident analysis codes results are important for the regulatory analyses required by the backfit rule (10 CFR50.109). Severe accident analysis codes were important in deciding whether to revise the regulations on steam generator tube integrity. Severe accident analysis codes have been important to the certification of the advanced and evolutionary light water reactor designs.

**7. What are your views on further development and maintenance of severe accident codes? Should multiple codes at different levels of detail continue to be developed/maintained for in-depth modeling and analysis of severe accident phenomena? Or should the codes be consolidated into a single integral code for use in plant safety and risk analyses?**

I believe that the NRC is in the business of regulating fission products. Since this is their business, the NRC needs the very best information about fission products it can possible get. I would think that the NRC would have very sophisticated models of fission product behavior under all circumstances - normal operations, upset conditions, and severe accidents.

But, given the realities of the situation, I think the consolidation to one model is a good thing if it is done well. For instance, I don't think that consolidation to MELCOR is a good idea until the specialized codes on fission product release and transport have been well validated against the data of the PHEBUS tests, the KAEVER tests and the forthcoming tests of fission product transport through the reactor coolant system. Consolidation on core degradation is much more palatable, just as has been consolidation on hydrogen combustion and core debris - concrete interactions.

I believe that real decisions about the continued development of codes await a decision by the Commission on the metric they want to use for risk-informed regulation. The Commission has set safety goals based on the language of risk which would require reliable methods for calculating severe accident phenomena. On the other hand, the implementation of risk informed regulation is based largely on core damage frequency. This does not require sophisticated accident analysis models. Is this reliance on core damage frequency an interim step and once the codes are better we will move to risk as the metric for risk informed regulation? Or, will we be stuck using core damage frequency as the metric forever?