NRC 19th Annual Regulatory Information Conference
March 13-15, 2007
Bethesda North Marriott Conference Center

Session Title: PRA Models, Methods, & Tools
Tuesday, March 13, 2007

Question: How are "Shared Systems" across units at a multi-unit site treated in the PRA?
Answer: Systems that are shared between units at a multi-unit site are treated in PRAs in various ways. First, assuming they are either front-line or support safety systems, they are typically taken credit for in preventing an accident at either unit. Conversely, particular attention should be paid to the special circumstances surrounding their dependencies. Shared systems may be subject to physical dependencies arising from the external events. This is because an external event, such as a flood, could disable mitigating systems at both units. There are also Human Reliability Analysis dependencies. For instance, the repair, maintenance, or testing regime may be the same for all shared systems at a particular site, which might contribute to their common mode failure even though they are physically separate between units and may even have been constructed at different times using different manufacturers. Finally, shared systems may exhibit their own functional dependencies which may arise if they rely on other shared support systems, e.g. diesel generators, to operate. Shared dependencies, such as these, should be modeled in the fault trees for each unit.

Question: Does Harmonization of Models Increase the Risk of Completeness Errors?
Answer: The word "harmonization" will be understood to be synonymous with "standardization." The standardization of models does not necessarily increase the risk of completeness errors. Consider the case of an auxiliary feedwater system and a fault tree model to estimate its reliability. One can never assure completeness. However, one must make every effort, when modeling the system, to incorporate all relevant knowledge. As more operating experience is gained, and more failure modes are included, completeness becomes more and more assured.
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NRC 19th Annual Regulatory Information Conference
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Session Title: New Reactor Organization and Applications: Status and Plans
Tuesday, March 13, 2007

Question: The Department of Homeland Security is required to make a determination of “reasonable assurance” that off-site emergency can protect the public. What criteria will DHS use to make this finding for a combined license application?

Answer: As established in 10 CFR 50.47 and 52.97, and Section 657 of the 2005 Energy Policy Act, the Nuclear Regulatory Commission (NRC) relies on the Department of Homeland Security (DHS) / Federal Emergency Management Agency (FEMA), for input to licensing reviews prior to issuing a new reactor license. Specifically, 10 CFR 50.47 (a) states that no initial operating license will be issued by the NRC unless a finding is made that adequate protective measures can and will be taken in the event of a radiological emergency. In addition, 10 CFR 50.47(a) states that the NRC will base its finding on a review of the FEMA findings and determinations as to whether State and local plans are adequate and whether they can be implemented. The regulations in 10 CFR 50.47(a) go on to state that a FEMA finding will primarily be based on a review of the plans. 10 CFR 50.47(a) also states that any other information already available to FEMA may be considered in assessing whether there is reasonable assurance that plans can and will be implemented.

DHS /FEMA acceptance criteria are provided in 44 CFR Parts 350, 351, and 352, including applicable DHS/FEMA policies, Radiological Emergency Preparedness-series guidance documents and associated memoranda, as they relate to off-site radiological emergency planning and preparedness.

Question: What impact, if any, do you anticipate from the application of DOE's Standby Support Rule?

Answer: As written the DOE's Standby Support Coverage proposal is satisfactory. We not anticipate any changes based on the existing understandings.

Question: How many COL applications do you anticipate by 2007? by 2008?

Answer: Recently the majority of prospective applicants provided response to an NRC Regulatory Information Summary requesting information of prospective applicants' plans. Not all the responses have been made public. We expect the majority of combined license applications to be submitted in the 2008 Federal fiscal year.

Question: How many will pursue getting a Standby Support agreement?

Answer: The DOE Standby Support Coverage is limited to the first six plants that start construction, defined as being the first pour of safety-related concrete. As a result we expect the majority of plants that apply in the 2008 fiscal year to apply for the coverage even though only six plants will qualify.

Question: To what extent have the various developer groups made requests for transmission service? Given long lead times for transmission siting, certification and construction, it could wind up being in the critical path.

Answer: Transmission from or to the site is not expected to be the critical path item. Applicants need to take transmission into consideration when preparing a combined license application. Discussions have already been held with transmission and distribution sections of the vertically integrated utility companies and with non-parent company transmission companies and organizations. To date the critical path items are not linked to transmission and distribution.

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NRC 19th Annual Regulatory Information Conference
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Session Title: Contaminated Groundwater and Lessons Learned Task Force
Tuesday, March 13, 2007

As mentioned, tritium in groundwater is not a public health and safety issue. What are we doing to balance 'real' risk versus 'perceived' risk? In some ways it seems we are responding, both in terms of licensee efforts and potential regulation, as if the perceived risk is real?

The NRC has determined that there has not been any public health impact from the ground water leaks, and has publicized this fact in the Liquid Radioactive Release Lessons Learned Task Force final report. The NRC has also stated this in public forums such as public meetings, the NEI Ground Water Lessons Learned Workshop, on the NRC public website, and here at the RIC. Thus the NRC recognizes that the actual risk is well below any public health impact and well below the 10 CFR 50 Appendix I ALARA criteria.

However, as you note, the public concerns that have been raised are based on their perceived risk of ground water contamination. In particular, the public has been concerned that there have been radioactive leaks that were unplanned, unmonitored, and in some cases undetected for relatively long periods of time. The NRC has provided public information on the unplanned releases on the NRC web site to include frequently asked questions, and a fact sheet on tritium.

See http://www.nrc.gov/reactors/operating/ops-experience/grndwtr-contam-tritium.html.

In addition, to ensure the NRC mission is being achieved to protect the public health and safety, the NRC are reviewing the existing regulatory basis and regulatory guidance for monitoring unplanned releases to ensure that the licensees have minimized the potential for further ground water leaks, by actively detecting, evaluating, and monitoring releases via unmonitored pathways.

As presented at the RIC, the industry has recognized the importance of maintaining public confidence in nuclear power plant operators, and therefore went forward with the industry Groundwater Protection Initiative (GPI).

Could the groundwater contamination issue lead to delay in new plant construction? If so, how will the NRC deal with that?

The NRC does not for see this issue causing a delay in new plant construction. Inadvertent releases of radioactive liquids have not resulted in any public health impacts.

Ralph Andersen's slide listed communicating to NRC, state, and local for <u>all</u> inadvertent releases. Is the current position that an onsite spill (> 100 gallons) meets the criteria of 10 CFR 50.72 and therefore needs to be reported to NRC? The slide listed communication to the NRC, State, and Local by the end of the next business day.

NRC guidance on making 10 CFR 50.72 reports is given in NUREG-1022, Section 3.2.12.
http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/

The NUREG states that the purpose of the 10 CFR 50.72 criterions is to ensure the NRC is made aware of issues that "will cause heightened public or government concern related to the radiological health and safety of the public or on-site personnel or protection of the environment." Therefore, if licensee communications to state and local authorities are made that are likely to cause heightened public or governmental concern, then that communication should be reported to the NRC as a 10 CFR 50.72 report. The NRC also notes that the 100 gallon threshold is an industry selected value, not the NRC threshold for reporting.

Are the enhanced reporting thresholds for on-site contamination? Is this just for Exelon or for public disclosure? Does it include nuclides other than tritium?

Since this is an industry established threshold, NEI should be contacted for further information. However, the NRC understanding is that these thresholds apply to both on-site and off-site, apply to all power plants, and include all radionuclide's.

What was the original detection method at Braidwood? What process caught it? Why didn't it catch it sooner?

The process was sampling done in preparation for NPDES permit renewal. Details are contained in NRC inspection report 05000456/2006008 dated May 25, 2006. This report is available to the public via ADAMS at ML061450522.

How did you get the name of groundwater users? Do most plants have such information?

Power plant licensees are aware of the communities and land use in the vicinity of the plant.
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NRC 19th Annual Regulatory Information Conference
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Session Title: Spent Fuel Storage and Transportation
Wednesday, March 14, 2007

Question: Will TADs be available for either rail or highway legal transport (i.e., the "mostly rail" scheme still intact).

Answer: TADs designed in accordance with DOE's Performance Specification will require the use of rail for cross country transport. There are no plans for developing a TAD canister system suitable for shipment by legal weight truck. Spent nuclear fuel shipped to Yucca Mountain in truck transportation casks will be received in the Repository's Wet Handling Facility and will be placed in TAD-based waste packages.

Question: How many TAD designs does DOE expect to select?

Answer: DOE has not made any determination as to the number of TAD designs that may ultimately be licensed for use at Yucca Mountain.

Question: When would the proof-of-concept designs be completed?

Answer: Work on the TAD Proof of Concept designs was completed at the end of February.

Question: The TAD spec precludes pyrophoric materials. Is any consideration of use of materials such as depleted uranium, which is not pyrophoric in a solid from, buy only as powder or shavings?

Answer: The exclusion of pyrophoric materials in the TAD Performance Specification is consistent with the Yucca Mountain project's approach to eliminating exothermic reactions anywhere in the engineered barrier system.

Question: You said TADs would be available to utilities in 2011. Is DOE expecting utilities to procure TADs prior to final approval of Part 63, 72, and 71 applications? Most utilities decide to transition to new dry storage systems 3-4 years before loading.

Answer: The Department plans to have vendors submit Safety Analysis Reports for storage and transportation to the NRC by the middle of next year. After review and approval by the NRC, these TAD systems should be available for utility use by 2011. The Department will encourage utilities to follow the TAD licensing process and utilize TADs for onsite storage as soon as practicable.

Question: Who does DOE expect to cover costs for transferring spent nuclear fuel from dry storage casks to TADs? If utilities are expected to cover these costs what is the basis for this opinion?

Answer: The Department does not have a contractual obligation to accept for disposal spent nuclear fuel contained within a dual purpose storage canister. Absent a mutually agreeable contract modification, utilities will have to repackage this fuel into a DOE supplied transportation cask.

Question: What performance specifications promote a low or a decreased vulnerability for the TAD canister system during storage and aging?

Answer: The TAD Performance Specification imposes additional requirements on the spent fuel aging system to be used at Yucca Mountain. These additional requirements address certain site specific natural phenomena that are not normally considered for the storage of spent nuclear fuel.

Question: Are the states in the survey requesting additional point-of-origin inspections beyond the currently required CVSA (I believe this stands for Commercial Vehicle Safety Alliance) level 4 (Earl believes this should be level 6 not level 4) inspections? For security escorts, are the states requesting additional security beyond the currently required armed security?

Answer: States in the survey expect to be conducting point of origin inspections in accordance with the commercial vehicle safety alliance (CVSA) North American standard at the level VI. I don't think States are anticipating multiple point of origin inspections but expect their own transportation authorities to actually do the level VI inspections. If CVSA level VI inspections are conducted at the point of origin, it is likely that some level of reciprocity will exist and that some states will honor the decal and not require additional inspections at state borders.

However, in some states, the border inspection is required by state statute. In these cases, some level of inspection, possibly a level II, will take place regardless of the CVSA level VI performed at the origin. Nearly the same situation exists for security. That is, for those states wishing to participate in security, State Police are willing to function as the “security escort” required by regulation. If the security or escort activity is required by state statute, they are likely to be present, even if additional security is provided by the shipper or carrier.

Question: Does every state want to conduct its own inspection as the waste package crosses borders or will point-of-origin inspections by all states be ok?

Answer: I think I attempted to answer this as part of the first question. For states that have inspection requirements by statute, some form of inspection will take place at the state borders. If a full level VI inspection was done at the point of origin, by a competent authority and reported to states downstream, it is likely that subsequent border inspections for a particular shipment would be an abbreviated.

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Session Title: Safety Margin Work
Wednesday, March 14, 2007

Question: Establishing a protective system to address a potential aircraft impact could be handled through defensive systems rather than over-engineering containment, thus being more cost effective. But may raise other issues with making each plant one step closer to a military type site.

- cost of over-engineering does need to be factored into the regulatory process
- probability x consequence = risks
- over-engineering containment to withstand an aircraft impact may be cost prohibitive

Answer: For a more complete answer, please refer to NUREG/BR-0314: “Protecting Our Nation - Since 9-11-01,” or visit the NRC web site at: http://www.nrc.gov/security/faq-security-assess-nuc-pwr-plants.html.

In keeping with its long standing philosophy, the NRC addresses potential aircraft impact through defense in depth. The efforts over the past number of years has been to use a combination of deterministic design basis analysis and risk studies.

In order to determine how much physical protection is enough, the NRC monitors intelligence sources to keep abreast of foreign and domestic events and remains aware of the capabilities of potential adversaries. The NRC uses this information, and other sources, to determine the physical protection requirements of its regulations and to establish and maintain design basis threats (DBTs). Nuclear power plants and selected fuel cycle facilities must be defended against the DBTs. The DBTs may be found in NRC regulation 10 CFR 73.1.

As part of a comprehensive review of security for NRC-licensed facilities, the NRC conducted detailed site-specific engineering studies of a limited number of nuclear power plants to assess potential vulnerabilities of deliberate attacks involving large commercial aircraft. In conducting these studies, the NRC drew on national experts from several DOE laboratories using state-of-the-art structural and fire analyses. The agency also enhanced its ability to predict accident progression and radiological consequences realistically. For the facilities analyzed, the vulnerability studies confirm that the likelihood of both damaging the reactor core and releasing radioactivity that could affect public health and safety is low. Even in the unlikely event of a radiological release due to terrorist use of a large aircraft, there would be time to implement mitigating actions and offsite emergency plans such that the NRC’s emergency planning basis remains valid. Additional site-specific studies of operating nuclear power plants are underway or being planned to determine the need, if any, for additional mitigating capability on a site-specific basis.

The Nation's nuclear power plants had implemented strong physical protection programs decades before September 11, 2001. The plants were already surrounded by fences with continuously monitored perimeter detection and surveillance systems, and they were guarded by well-trained and well-armed security forces. The plants also have redundant and diverse safety equipment so that if any active component becomes unavailable, another component or system will satisfy its function. In addition, plant operators had been trained to respond to
unusual events and emergencies, and each plant has carefully designed emergency plans in place
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NRC 19th Annual Regulatory Information Conference
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Session Title: Proposed Geological Repository at Yucca Mountain
Wednesday, March 14, 2007

Question to Ward Sproat, DOE: What is the status of DOE preparations for work on standard contracts for new reactors?

Answer: The work on a standard contract for new reactors is underway. The department expects to provide its perspective on this issue later this summer.

Question to Ward Sproat, DOE: What is the status of contract awards for remaining two independent reviews?

Answer: The contracts for the independent review of the License Application and the independent review of quality assurance are expected to be awarded in April 2007.

Question: If DOE is sitting on $19 billion (with several $100 million going in per year); why not get legislation to allow DOE to use this funding now for Yucca preparation?

Answer: The Administration's proposed legislation does have funding provisions which would allow greater access to the yearly fees from utilities for Yucca Mountain.

Question to Ward Sproat, DOE: Section 7 of 5.2589 and the new "Save Yucca" legislation is counterproductive to development of a transportation system that includes buy-in and confidence of corridor states. Why?

Answer: The proposed legislation does not displace existing Federal, State, local or Tribal requirements; rather, it permits the Secretary of Energy to request that the Secretary of Transportation use the existing administrative process under the Hazardous Materials Transportation Act to resolve specific efforts to obstruct shipments to Yucca Mountain.

DOE will continue to cooperate with Federal, State, local and Tribal entities to use existing standards, expertise, and resources to the extent practical in developing safe and secure transportation plans. DOE will also continue its regular consultations with State, local and Tribal entities and the provision of financial and other assistance to these entities, as appropriate.

Question: How would the nuclear industry proceed with its integrated used fuel management plan, if work on the Yucca Mtn repository were to be terminated?

Answer: Termination of Yucca Mountain seems a particularly severe outcome given all the options available to make it successful and the continuing bipartisan support for the project in Congress.

However, if such should come about, the industry would first continue doing what it has been doing since the beginning – storing used fuel on our sites safely and securely. The industry would continue to advocate for advanced fuel cycle development with interim storage ideally at a location associated with such developmental facilities.

Any termination of the Yucca Mountain project would not relieve the federal government of its contractual obligation to remove used fuel from commercial sites.

Regarding a repository, regardless of the fuel cycle employed, disposal of a by-product will be needed. It has been long established as national policy that the federal government is responsible for ultimate disposal of used nuclear fuel and/or the by-products from fuel processing. The industry, along with the rest of the world, would look to Congress to deal with this national question.

Question: How can you honestly argue that Yucca should "fail" because performance in one <u>million years</u> cannot be proven, but that on-site storage is "viable" because GBCS are good for <u>hundreds</u> of years?

Answer: It remains to be seen whether DOE will be able to credibly assess the total system performance of the natural and engineered barriers of Yucca Mountain over one million years because, among other things, the site geology is complex, important data is missing, and the performance of the engineered barriers depends on complex and interrelated thermal, chemical, and hydrological factors. Assessing the relatively short-term performance (hundreds of years) of SNF storage casks located on already reviewed and approved nuclear plant sites is simpler.

Question: Although the U.S. Nuclear Regulatory Commission Chairman may be able to decouple new reactor construction from a permanent repository, the local communities cannot. What would you say to the local government leaders and residents of communities who believe their communities have become de facto permanent repositories?

Answer: Reactor sites will not become de-facto permanent repositories because the federal government recognizes it has a legal obligation to accept SNF for disposal elsewhere. However, one should be candid and admit to communities and local governments that Yucca Mountain has not been proven to be safe, and that SNF may need be stored on the reactor site for a very long time, certainly many decades, before it can or will be moved elsewhere. This is a downside for nuclear power that has to be weighed against its many benefits.

Question: How do you respond to accusations of NIMBY on the part of the State Nevada?

Answer: Nevada will raise significant safety and environmental questions about Yucca Mountain, not frivolous ones. Someone who raises significant questions is not suffering from a NIMBY syndrome, but is instead exercising its rights of responsible citizenship and sovereignty.

Question: Please give us your perspective on their risk significance of existing EPA standards on radon, as compared to two 350 m rem limit in the draft rule.

Answer: By law, EPA's proposed Yucca Mountain rule must establish an acceptable level of risk, without regard for feasibility or economic costs. Such a standard is sometimes called a health-based standard, one that is fully protective of human health. Congress has not required EPA to issue a comparable standard that defines an acceptable level of risk from radon in indoor air. Instead, for radon in indoor air, EPA has purely voluntary guidelines that recommend reduction in levels above 2pCi/L, noting that reduction below 2pCi/L would be infeasible. EPA does not assert that a level at or below 2 pCi/L is fully protective of human health, just that it represents a feasible risk reduction goal.

Question: What makes you believe, given DOE's performance on Yucca Mtn, that DOE would be capable of operating and maintaining the ISFSI's in the U.S.?

Answer: This is a legitimate question, but DOE should be able to draw on the experience and expertise of the nuclear industry to assist it. Moreover, operating a dry cask storage facility is much simpler that assessing the safety of Yucca Mountain.
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Session Title: 10CFR 50.46 and Acceptance Critieria
Wednesday, March 14, 2007

<u>Question:</u> In preparing the proposed 50.46a rule, the staff met with the ACRS (full and subcommittee) on 6 different occasions. The expert elicitation process and results were discussed a number of times. Is the ACRS fulfilling its role by raising concerns with the expert elicitation when the staff presents the rule for final review by the ACRS when in the previous meetings, similar concerns were not raised?

<u>Answer:</u> One must realize that ACRS is a dynamic organization with constantly changing membership. Any recent concerns about the expert elicitation process were raised by new members not present during the earlier meetings. My impression, however, of the ACRS view of the expert elicitation process is that it was very well done and that the ACRS had (and still has) no significant concerns about it and the results. The "new" issue was that, in general, when a very important decision is to be made as a result of findings from an expert elicitation process, the process should be subject to a peer review. The ACRS reviews should not be viewed as peer reviews.

<u>Question:</u> How can the rule objectives stated in Mr. Dudley's presentation be accomplished without revising the current rule?

Objectives:
- Focus NRC and Licensee resources on more risk significant issues
- Enable licensees to enhance overall safety by modifying plant designs to increase design emphasis on smaller breaks.

<u>Answer:</u> My objection to the rule change, in essence, is that when one cannot pre-determine in advance the risk implications of altering a Design Basis Accident description, then it is best not to do it. The NRC staff recognized that it is basically impossible to determine the risk impact of such a change, so they put constraints on the change such that every resulting change is tracked and the risk increase kept to a very small value. My contention was that this tracking and maintaining a small risk increase amounts to the same thing as the process in R.G. 1.174. I do not think the efficiencies afforded by the rule change are worth the potential for risk increases that go unreviewed by the staff. In addition, this "meddling" with DBA space without a methodology to pre-determine the risk impact sets a precedent that should not be continued.

<u>Question:</u> Since GDC #4 de-couples LBLoca analysis from the actual design for dynamic loads, what is the value of providing extra "margin" in the FSAR LBLoca analysis?

<u>Answer:</u> There is more to margin than just its effect on the dynamic loads. Nevertheless, the ACRS did not advocate extra margin – just the preservation of margins already afforded by the DBA prescription and the preservation of defense-in-depth both of which are mandated by the R.G. 1.174 licensing change process.

<u>Question:</u> Redefinition of the DBLOCA would mean that breaks greater than the Transition Break Size would be beyond design basis accidents. If this is an acceptable premise for the rule than why is the ACRS suggesting that additional controls are necessary for breaks greater than the TBS which would be beyond design basis.

Answer: The ACRS does not view the concepts of Design Basis Accidents and the spectrum of real accidents that include those euphemistically called “beyond design basis” as similar things that exist on a continuum. The DBAs represent a construct that is hopefully used to render a design to an acceptable level of risk. Unfortunately, there is not a one-to-one correspondence between design basis space and risk space. We do not have a methodology for assessing the real risk impact of any changes to the DBA specifications. Therefore, the ACRS considers the additional constraints placed on this to be a prudent way to assure that the risk impacts are probably acceptable and that margins and defense-in-depth are sufficient.

Question: The ACRS was critical of the staff’s proposed rule from the standpoint that it did not specify special treatment and control for equipment needed to mitigate breaks larger than the transition break size. This seems inconsistent with the Commissions direction to the staff which noted that there should not be overly prescriptive regulatory treatment of beyond design LOCAs.

If, for example, the staff were to use 10CFR 50.67 to categorize equipment needed to mitigate breaks greater than the TBS, it would show these systems to be of low safety significant and would not warrant special treatment. This appears to be an inconsistent use of risk. Please comment.

Answer: I was tempted to quote “foolish consistency is ......”. However, the real answer is that ACRS is concerned primarily with risk impacts of any messing around with design basis space.
See the answer to the above question of their view vis-à-vis DBAs and “beyond DBAs” as an artificial construct that cannot be translated into real risk impacts. On the other hand, we do not have a methodology or a data base that can be used to judge the risk “reduction” due to special treatment requirements. It is merely a judgement that special treatment is worth the burden. There are members of ACRS that do not think it worth it even for SSCs. It has yet to be shown to us that the mitigation capability for LBLocas should not be rendered as an SSC.
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NRC 19th Annual Regulatory Information Conference
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Session Title: Construction Inspection Program
Wednesday, March 14, 2007

Question: What activities or programs are in progress to assure that there are appropriate codes and standards developed for the new designs (ASME/IEEE/ANSI)?

Answer: The NRC staff is currently reviewing and updating the Standard Review Plan and Regulatory Guides. These documents typically reference the appropriate national codes and standards that are acceptable to meet portions of the NRC regulations. In addition, as part of the licensing review process for the combined license, the NRC staff will review the codes and standards that the applicant commits to meet in the license application and determine if the regulations are met. Many NRC staff members participate in codes and standards committees to ensure these documents remain up-to-date.

Question: INPO is in the early states of development of a construction evaluation program. Will there be an effort to collaborate with the NRC to avoid duplication - understanding both organizations’ need to maintain independence?

Answer: The NRC has met with INPO to provide an overview of the construction inspection program and to understand the role of INPO. INPO has assigned staff to work with industry to establish their role in the construction of new plants. In the past, INPO has conducted evaluations of construction activities and operational readiness reviews prior to plant startup. As with the operating reactors, NRC will coordinate its activities at the sites with the licensees and INPO to ensure no unnecessary burden on the licensee. However, the NRC's construction

inspection program will be conducted as described in its Inspection Manual, and the INPO activities will not take the place of any NRC inspections.
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NRC 19th Annual Regulatory Information Conference
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Session Title: Rulemaking
Wednesday, March 14, 2007

Question to Joe Baurer of Exelon Nuclear: How does the industry task force get buy-in from all the sites?
Answer: Regarding process for implementation guidance, particularly those who are not participated in the task force, the licensee representatives who participate on the NEI Task Force jointly arrived at the industry position for various issues addressed in the NEI guidance document. I am not aware that other licensees (not participating on the Task Force) have been engaged.

Question: What is the status of the NEI implementation guidance on fatigue?

Answer: The NEI Task Force is scheduled to meet again on 03/20/07 to continue working on the guidance document. We anticipate having the document 95% completed pending potential changes in the final rule language.

Question: Were discussions with the trade unions held as part of the rulemaking for work hour rules and if so what concerns/issues were identified?

Answer: The IBEW was in attendance at a number of the public meetings held to discuss the rule. In general, the IBEW representative expressed concerns to the NRC that the new rules were overly restrictive and would negatively impact the workforce. There also has been some discussion between vendor/contractor organizations and the NRC in recent months. The contractor organizations expressed concerns that the new restrictions on work hour during outages may prompt craft workers to avoid working in the nuclear industry in preference to other less regulated industries where overtime is plentiful.

Question to Jack Roe of Nuclear Energy Institute (NEI): How does industry decide when to engage the NRC in rulemaking with respect to implementation?

Answer: The industry will address implementation based upon NRC's progress in developing the guidance.
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Session Title: Reactor Inspection and Assessment
Wednesday, March 14, 2007

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1. During the regions mid-cycle and end of cycle meeting, what discussion takes place, when at least 2 findings with the same aspect are “identified”?

If the cross-cutting area is human performance or problem identification and resolution a discussion takes place if there are more than three current inspection findings with the same cross-cutting aspects and a common theme which may result in a substantive cross-cutting issue, if other criteria are also met. Hence, if only two findings with the same cross-cutting aspect are identified in the human performance or PI&R areas, the licensee has not exceeded the criteria for a substantive cross-cutting issue and the findings are therefore not discussed

However, in the safety conscious work environment (SCWE) cross-cutting area, a substantive cross-cutting issue requires at least one finding or the NRC has issued SCWE-related correspondence, as well as meeting other criteria, and this would be discussed during the mid or end of cycle meeting.

2. How does the NRC keep inspectors and the regions from “regulating to excellence” when deciding if a cross-cutting aspect or theme is present?

Findings are evaluated to determine if there is a licensee performance deficiency. The performance deficiency has to be an issue that is the result of a licensee not meeting a requirement or standard where the cause was reasonably within the licensee’s ability to foresee and correct, and that should have been prevented. A performance deficiency can exist if a licensee fails to meet a self-imposed standard or a standard required by regulation. Findings are not written for issues where a licensee fails to meet an industry practice to achieve excellence. Since these issues (involving a failure to meet an industry practice to achieve excellence) are not findings they do not result in a cross-cutting aspect or theme.

3. Many utilities define error prevention tools outside of procedures. How does the regulatory process support identifying cross-cutting aspects for a voluntarily implemented standard?

As defined by Inspection Manual Chapter 0612, "Power Reactor Inspection Reports" a Performance Deficiency is an issue that is the result of a licensee not meeting a requirement or standard where the cause was reasonably within the licensee’s ability to foresee and correct, and that should have been prevented. A performance deficiency can exist if a licensee fails to meet a self-imposed standard or a standard required by regulation.

If the performance deficiency has related cross-cutting aspects, the cross-cutting aspects are considered a significant underlying cause of the performance deficiency rather than an independent issue. Issues of problem identification and resolution, human performance, or establishment of a safety-conscious work environment, in and of themselves, do not provide the basis for a performance deficiency.

Findings are NRC-identified or self-revealing issues of concern that are associated with a licensee performance deficiency. For findings of greater than minor significance, the criterion in IMC-0612 is utilized to determine whether a cross-cutting aspect is associated with a finding. In this manner the regulatory process supports identification of cross-cutting aspects for a standard that is voluntarily implemented by a licensee.

4. Under what circumstances would the NRC disagree with the licensee's root cause for events that are potentially cross-cutting issues? Provide examples.

Inspection Procedure (IP) 71152, "Identification and Resolution of Problems" provides the primary mechanism for NRC inspectors to examine the adequacy of licensee corrective actions and the associated apparent and root cause analyses. IP 71152 references IP 95001, "Inspection for One or Two White Inputs in a Strategic Performance Area." IP 95001 provides extensive guidance for NRC inspectors to use when they evaluate the adequacy of a licensee's root cause analysis. If the licensee's evaluation failed to adequately evaluate the extent of condition or extent of cause of the issue, the NRC could then take issue with the adequacy of the licensee's analysis.

While not directly involving a potential cross-cutting issue, an illustrative example is provided for a site where a supplemental inspection was conducted in accordance with Inspection Procedure 95001, to assess the licensee's evaluation associated with; (1) the performance indicator for excessive safety system unavailability for the heat removal system (due to a degraded auxiliary feedwater pump) crossing the threshold from Green (very low risk significance) to White (low to moderate risk significance) for the site in the fourth quarter of 2005, and (2) the White finding for the auxiliary feedwater pump B being out of service for greater than the technical specification allowed outage time due to an incorrectly installed bearing and subsequent inadequate corrective actions. Specifically, the site shared "B" turbine driven auxiliary feedwater pump was discovered in a degraded condition on November 7, 2005. The licensee determined the pump had an incorrectly installed bearing which resulted in inadequate lubrication of the inboard pump bearing. The pump was determined to be inoperable and unable to meet its expected mission time from December 14, 2004 until November 11, 2005.

The licensee's problem identification, root cause and extent-of-condition evaluations, and corrective actions for the degraded pump were generally adequate. However, several deficiencies were identified by the inspector relating to the thoroughness and quality of the root cause evaluation and subsequent corrective actions. Of note, the root cause evaluation did not identify that an evaluation required by the ASME code was not completed when the auxiliary feedwater pump B was returned to service with high vibrations on September 3, 2003. Therefore, the White finding will remain open pending development of corrective actions to address these NRC-identified weaknesses.

5. IP's 95001, 95002, and 95003 now review the licensee's evaluation of safety culture components impact on the causes of the finding. Can you comment on the NRC's view of licensees' performance in this area since the change in the IP's and have the additional evaluations being performed for greater than green findings improved the license's corrective actions compared to evaluations prior to the incorporation of safety culture into the ROP?

The NRC is in an 18-month initial implementation period for the enhanced ROP that began on July 1, 2006. We do not have any data at this time that shows whether the revised supplemental inspection procedures have had a direct impact on the quality of licensee corrective actions.

6. What is the mission and goals of the NEI ROP Task force?

To gain an understanding of the mission and goals of the NEI ROP Task Force please contact NEI.

7. You stated safety culture was transparent, understandable, predictable, etc. But in fact the implementation has identified that there is a lot of subjectivity and is resulting in differences across regions. This is also impacted by minimal training for inspectors (based on inspector admission). What and when will adjustment be made to return to predictable? Have we actually reverted back to pre ROP subjective approaches?

No, the changes that were made related to safety culture did not significantly change the way the NRC assesses plant performance. All four regions conducted the end-of-cycle assessments in accordance with Inspection Manual Chapter 0305, "Operating Reactor Assessment Program." The Manual Chapter was consistently applied by all four regions. The outcome of the individual plant discussions were different based on specific plant performance over the last year, but that is expected. For instance, the number of plants in elevated columns of the Reactor Oversight Process (ROP) Action Matrix, that is the Regulatory Response Column and above, are 5 for Region I, 12 for Region II, 8 for Region III, and 5 for Region IV. However, as discussed in the Region IV breakout session, the number of plants with substantive cross-cutting issues is highest in Region IV (5 sites). As discussed in the Region IV breakout session, the NRC will be reviewing the results from the end-of-cycle assessments as part of an effort to review substantive cross-cutting issue consistency.

8. IAEA decided not to have a guide on safety culture. Instead they developed a requirement (NS-R-3) and guides on Management Systems. How does NRC react on these developments?

The NRC is aware of the IAEA safety culture activities. NRC staff review IAEA draft guides and other technical reports, as appropriate. The NRC's safety culture initiative considered IAEA's safety culture attributes and characteristics in developing the NRC's safety culture components. NRC staff recently attended an IAEA Technical Meeting on Integrated Management Systems which incorporate considerations of safety culture in IAEA's GS-R-3 and Draft Safety Guide DS349. At this time there are no plans to modify the NRC's approach toward safety culture.

9. Is a safety managements system a suitable tool to overcome or minimize safety culture problems?

IAEA is currently developing documents that address integrated management systems which incorporate considerations of safety culture as a way for licensee's to ensure a safety culture. Currently, several licensees abroad are implementing integrated management systems. There are a variety of regulatory strategies and processes that can conceptually be employed to enhance licensee safety culture. A safety management system is one type of suitable approach.

10. Does the constraint not to fundamentally modify the ROP structure diminish the perceived importance of safety culture?

No, safety culture is an important aspect of licensee performance. The changes made to the ROP were intended to provide: (1) better opportunities for the NRC staff to consider safety culture weaknesses and to encourage licensees to take appropriate actions before significant performance degradation occurs, (2) the staff with a process to determine the need to specifically evaluate a licensee's safety culture after performance problems have resulted in the placement of a licensee in the degraded cornerstone column of the action matrix, and (3) the NRC staff with a structured process to evaluate the licensee's safety culture assessment and to independently conduct a safety culture assessment for a licensee in the multiple/repetitive degraded cornerstone column of the action matrix. By utilizing the existing ROP framework, the NRC believes its oversight activities are based on a graded approach and remain transparent, understandable, objective, risk-informed, performance-based, and predictable.

11. What formal input/review has the industry or NRC have been done by behavioral specialists to assure that safety culture initiatives do not have unintended consequences?

The ROP enhancements to address safety culture were developed by NRC staff with highly specialized expertise in organizational behavior. In addition, the ROP enhancements were discussed and reviewed by a wide cross-section of external stakeholders during the developmental process. Comments made by external stakeholders on the Inspection Manual chapters (IMCs) and Inspection Procedures (IPs) that were enhanced by the safety culture initiative were addressed by the staff and the evaluations posted on the safety culture public web-site at: http://www.nrc.gov/about-nrc/regulatory/enforcement/safety-culture.html#finalimc.

During the developmental process, the NRC staff initially screened the safety culture components, and further screened them with external stakeholders, for the potential for unintended consequences and deleted some safety culture components that were thought to have the potential to have unintended consequences. The initial 18-month implementation period will allow the opportunity to identify unintended consequences of the safety culture revisions to the ROP.

12. Could you give examples of findings where the inspector does not identify a cross-cutting aspect?

The staff's review of inspection finding information shows that approximately one-third of the inspection findings do not have a cross-cutting aspect. It is noted that findings with cross-cutting aspects must meet the following requirements: the finding is more than minor; the cross-cutting aspect was a significant contributor to the finding; the cross-cutting aspect is reflective of current licensee performance; and the cause of the finding is related to one of the three cross-cutting areas.

Some examples include:

A self-revealing finding was identified regarding the failure to install heat trace on the standby liquid control system in accordance with the vendor manual. The heat trace was installed in 1994 without the required ground-fault circuit protection. This resulted in a small fire in the heat trace on November 11, 2006. This issue was entered into the licensee's corrective action program as Condition Report CR-006-09006.

The team identified a finding for the failure to correctly translate the design basis of the containment ventilation backup nitrogen bottles into procedures. Specifically, the operator rounds log was revised to allow the two backup nitrogen bottles, which operate containment vent valve 1GSHV-4964, to decrease to 200 psig each. The nitrogen capacity calculation assumed a minimum of 800 psig per bottle to ensure sufficient nitrogen to stroke the containment vent valve as needed in beyond design basis events. Operation below 800 psig did not ensure the containment vent valve could be used according to emergency operating procedures to protect containment against overpressurization. The licensee raised the minimum backup nitrogen bottle pressure to 800 psig per bottle, performed a review of past bottle pressures, and initiated a notification to change the operator rounds log to allow a minimum of 800 psig per bottle. There were no violations of NRC requirements because the containment vent function is not covered by Technical Specifications, is not a part of the plant's licensing basis, and is only credited in beyond design basis events.

13. Should NRC or licensee ID significant safety culture deficiencies that the licensee does not satisfactorily resolve, what regulation(s) support enforcement? (E.g. "Contrary to ...reg/license condition)

Significant safety culture deficiencies that the NRC or licensee identifies should be entered into the licensee corrective action program. If the licensee does not resolve the issue, this would be identified in a PI&R inspection. The Licensee should also identify this through self-assessment and effectiveness reviews of the corrective action program. A performance deficiency can exist if a licensee fails to meet a self-imposed standard or a standard required by regulation. So enforcement would depend on the specifics of the finding.

14. One of the changes made to 0305 this year was to require that the cross-cutting aspect for a finding be directly related to the cause of the performance deficiency. As a result: 1. We cannot document a PI&R aspect for anything other than a criterion XVI violation, and 2. This has given the licensee's the impression that we are elevating cross-cutting aspects into separate findings. Is this what you intended?

For there to be a finding with a cross-cutting aspect, the cross-cutting aspect has to be a significant contributor to the inspection finding. Hence, the cross-cutting aspect is a significant contributor to the finding, and is not a separate finding. The PI&R cross-cutting area includes the following cross-cutting area components, corrective action program, operating experience, and self and independent assessments. Each component has several aspects. See IMC 0305 for all aspects. If any of the cross-cutting aspects identified in IMC 0305 are a significant contributor to the finding then that finding has that cross-cutting aspect. The cross-cutting aspect, which is a significant contributor to the finding, and the regulatory requirement that has been violated, such as Criterion XVI, are two distinct concepts. There is no intent to have all findings with PI&R cross-cutting aspects to be associated with Criterion XVI violations, although this is the regulatory requirement that is most closely aligned with typical findings involving inadequate corrective actions.

15. You indicated that the self assessment results from each licensee may be reviewed. How do you handle INPO assessment results?

Each regional office conducts a mid-cycle and end-of-cycle review utilizing the most recent quarterly performance indicators and inspection findings compiled over the previous twelve months. During these assessments, the NRC considers the conclusions of any independent assessments of a licensee, such as Institute of Nuclear Power Operations (INPO) and International Atomic Energy Agency (IAEA) Operational Safety Review Team (OSART) inspections. The purpose of considering independent assessments is to provide a means of self-assessing the NRC inspection and assessment process. References to INPO conclusions will not be included in the assessment letters.

16. Does it make sense that SCWE is a cross-cutting issue when safety culture is not? (When SCWE is a subset of safety Culture).

Yes, safety culture encompasses many aspects, including the three cross-cutting areas of human performance, PI&R, and SCWE. Since all three cross-cutting areas are a subset of Safety Culture, all of the findings associated with any one of those three cross-cutting areas, or the existence of a substantive cross-cutting issue in any of the cross-cutting areas, is indicative of a safety culture concern. Safety culture also encompasses accountability, continuous learning environment, organizational change management, and safety policies which are not associated with the cross-cutting areas. See Inspection Manual Chapter 0305, "Operating Reactor Assessment Program," for more details.

17. In regard to the objective of being "predictable", the threshold for a lower level issue (e.g. NCV) for being a CCI is not clear or predictable to the licensee. Please comment on this.

All findings with a cross-cutting aspect must meet the following requirements:

1. The finding is evaluated as more than minor (note: cross-cutting aspect of the finding will not be used to determine whether the finding is greater than minor)
2. Cross-cutting aspect was a significant contributor to the inspection finding
3. The cross-cutting aspect of the inspection finding is reflective of current licensee performance
4. Cause of the finding is related to one of the three cross-cutting areas (Problem Identification and Resolution, Human Performance, or Safety-Conscious Work Environment)

NCV's are "lower level issues", however they have been screened as more than minor. The process is "predictable" because all findings that are evaluated as more than minor will be considered for cross-cutting aspects. All findings that are minor will not be evaluated for cross-cutting aspects. The enhancement of the ROP cross-cutting aspects had no effect on the process to determine which findings are more than minor. If the significance of a finding is not clear to a licensee, they should interact with the regional office.

18. What do you mean by “we focus on continuous improvement?” How does the regulator achieve that with Reactor Inspection and Assessment?

The reference to continuous improvement was to NRC processes and not to licensee performance. We continue to improve the ROP by performing audits of inspection reports, performing effectiveness assessments of programs, soliciting feedback from stakeholders, and reviewing independent assessments. This helps ensure that the NRC is focusing its resources appropriately.
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NRC 19th Annual Regulatory Information Conference
March 13-15, 2007
Bethesda North Marriott Conference Center

Session Title: Severe Accident Research and Regulatory Applications
Thursday, March 15, 2007

Question (to C.Tinkler): You have said that you expect the SOARCA program will not evaluate the consequences of the SST1 source term. This will immediately reduce the peak consequences by at least an order of magnitude. What is the technical basis for this exclusion, and how do you expect the public to accept this without the suspicion that you are restricting the study so that it will yield the results you want?

Answer (Tinkler): The NRC presentation did not specifically address the SST1 radiological source term that was used in the 1982 siting study. However, to the point of the question, we do not plan to specifically evaluate the consequences of the SST1 source term simply because it has been used in past analyses. As we have stated, past studies utilized what are now known to be excessively conservative or non-credible assumptions and boundary conditions and do not reflect what has been learned from 25 years of phenomenological research worldwide. Further, past studies also do not reflect plant improvements or inherent features which mitigate against extreme source terms. The SOARCA program will use radiological source terms based on consistent, mechanistic analyses employing modeling (i.e., MELCOR code) that reflects the advancements in understanding and those plant improvements. Communication of the studies results and the reasons behind those results will thoroughly address how the source terms for this new evaluation were analyzed and their bases. We will also communicate how our improved understanding has led to changes in the source term, such as SST1, from the earlier attempts (e.g., 1982 siting study, NUREG/CR-2239) to characterize radiological source terms. Through a clear communication of the bases for this improved evaluation and description of what has changed from those earlier studies, we believe stakeholders will develop confidence that the SOARCA program represents an updated and superior characterization of plant risk and consequences and that changes in the outcomes are due to real improvements in understanding and plant performance over the last 25 years.

Question (to C.Tinkler): Dominant sequence and plant mitigation features are plant specific, how does SOARCA provide new insight beyond NUREG-1560?

Answer (Tinkler): NUREG-1560 is the NRC summary evaluation of the Individual Plant Examination (IPE) program, the NRC’s perspectives on reactor safety and plant performance gained from reviewing licensee submittals. While the IPE evaluations reflected significant improvements in evaluating plant specific core damage frequency and a more systematic evaluation of containment performance from earlier risk assessments; the IPE’s themselves are now dated and do not reflect the latest status of plant improvements and phenomenological understanding. Importantly, the IPE program did not include in its evaluation the offsite public health consequences of events and thus does not reflect improvements in emergency planning and the modeling thereof.

Question (to C.Tinkler): Since you are planning to incorporate dose response thresholds in the analyses, are you also going to consider the supra-linear dose response models that are suggested by recent findings in genomic instability effect?

Answer (Tinkler): The specific details of the treatment of effects of low doses of radiation which will be used in the SOARCA study have not been established. We stated our plan to convene a group of experts on the health effects of radiation to solicit their conclusions on this matter, including consideration of a dose threshold below which there are no observable adverse health effects or effects are non-existent. Based on the findings of that expert group a consistent technical approach will be adopted.

Question (to C.Tinkler): Do you have any plans to expand this work to include accidents at shutdown? If not, why not?

Answer (Tinkler): There are no plans to include an assessment of risk and consequences from accidents occurring at shutdown. This study is intended to update past studies of accidents occurring at power operation. [It is concluded that events proceeding from power operation pose the greater challenge in terms of the timing of potential fuel damage since those events are initiated at higher reactor decay power levels.]

Question (to C.Tinkler): Will the NRC use the results of SOARCA to reduce or increase the emergency plan zone around plants?

Answer (Tinkler): There are no plans to use the results of SOARCA for such purposes.

Question (to C.Tinkler): Why does the NRC not choose Indian Point for a bounding analysis using SOARCA?

Answer (Tinkler): The NRC has not determined all the plants which will be evaluated as part of the SOARCA study. The preliminary plan is to select plants representative of the variety of reactor and containment designs and to perform realistic evaluations of those selected plants.

Question (to C.Tinkler): With the possibility of plant life extension to 60 to 80 years, how is aging being considered in the SOARCA program?

Answer (Tinkler): There is no need for explicit treatment of aging in the SOARCA study. Licensee programs (and the NRC's research, regulations, and oversight activities) specifically evaluate and manage the issues which arise from plant aging. It is the objective of these aging management programs that safety margins inherent in the plant design be maintained. The NRC reviews and approves licensee plant extension applications to provide reasonable assurance that risk due to plant aging remains acceptable throughout the extended life of the plant.

Question (to C.Tinkler): How will the industry extensive work in L2/L3 [level 2/level 3 PRA] as part of license renewal be used to benefit the NRC SOARCA program?

Answer (Tinkler):It is our expectation that licensee activities related to license renewal will be an important source of information on dominant accident scenarios, plant improvements including new procedures, and a potentially source of information on more realistic, treatment of emergency planning. It is our expectation that the information in license renewal applications represents an update and improvement over the Individual Plant Examinations. We would expect to use any such licensee information to inform the selection of dominant sequences for NRC's analysis.

Question (to C.Tinkler): Will NRC make it clear that SOARCA supersedes the 1982 Sandia study? Schedule of SOARCA?

Answer (Tinkler): The documentation associated with SOARCA will communicate that these analyses supersede previous related NRC studies, including the 1982 Sandia study. The schedule for SOARCA currently calls for completion of the assessment of up to eight plants, representing the major reactor and containment design variations, in 2009.

Question (to C.Tinkler): Given TMI, Chernobyl, Davis Besse, and other significant events not modeled in PRAs, why can/should we concentrate on the “dominant” ones in level-1 PRAs? Will you model B.5.b improvements? Will you look at external events?

Answer (Tinkler): The citation of events which either; 1) have no specific relevance to US designs (Chernobyl), 2) predate the substantial commitment and activities to improve operator training, emergency operating procedures and severe accident studies (TMI), 3) have no generic impact on initiating event frequencies (Davis Besse), do not invalidate the use of level 1 PRA modeling to guide the identification of dominant or important accident sequences.

We plan on modeling all relevant plant improvements which would prevent or mitigate core damage and offsite radiological release. External events are being screened to determine if there is a correspondence, from a systems availability standpoint, to internal events. If there is a functional similarity between external events and certain internal events then the frequency of the events would be adjusted. External events are also being screened to determine if unique challenges are posed by the event itself.

Question (to Tinkler): Does MACCS code modeling take into account rain pattern’s effects? Is this an important consideration given Chernobyl lessons learned?

Answer (Tinkler): MACCS code modeling treats rainfall as one of the elements of weather conditions which may affect dispersal and deposition of radioactive material. In particular, MACCS has models for the increased deposition of radioactive material due to rain washing out that material from the atmosphere. The importance of rain related effects on the predicted offsite consequences depends on the combination of factors which influence radionuclide release, transport and deposition as well as modeling of offsite protective measures.

Question (to R. Gauntt): If TMI reactor core vessel had given way, would the containment have held? If no, does this challenge the delayed release assumption?

Answer (R. Gauntt): It should be emphasized that the delayed release characterization in this discussion is a calculated outcome from severe accident progression analysis using our state of the art codes (MELCOR) and is not an assumption of the SOARCA project. Had the TMI-2 vessel failed, the fraction of the reactor core that had melted would have entered the lower cavity where interactions with the cavity concrete would be initiated. While design specific features can affect the progression and effects of the core-concrete interactions and ultimately whether a containment failure eventually occurs, in general such interactions have the potential to fail the containment by long term static overpressurization from the concrete gaseous decomposition products (CO_2 principally). Such failures, if they occur, however, occur late in time, typically on the order of 24 - 72 hours. So, the prospect of vessel failure of attack from molten core materials does not itself challenge the delayed release characteristics of severe accident sequences.

Question (to R. Gauntt): How would you graph unreleased magnitude and duration if you looked only at LB-LOCA and didn't bundle it together with SBO + SB LOCA?

Answer (R. Gauntt): The cumulative distributions characterize a spectrum of analyses, roughly in proportion to their likelihood, and include the LB-LOCA contribution. If we included only the LBLOCA we would not have a distribution; rather, we would be showing a single point value, and this value would show shorter duration compared to others in the distribution, and larger volatile fission product release.

Question (to R. Gauntt): Bob Henry stated that MAAP analyses of the TMI-2 accident greatly overestimated structure temperature in the upper plenum, and underestimated the amount of energy remaining in the damage core. Could you characterize MELCORs prediction of the accident in these contests? Do you have any suggestions on how to improve the modeling?

Answer (R. Gauntt): We continue to explore these observations with our ongoing TMI-2 assessment work, but do not conclude that MELCOR is over-estimating upper plenum temperatures. Our modeling of core to upper plenum is assessed against the Westinghouse 1/7th scale experiments, and in this case produces good comparison to the available data. It also should be pointed out that the TMI-2 upper grid showed regions of localized melting in a non-symmetric pattern, which indicates that gas temperatures reaching the upper grid certainly exceeded the melting point of the grid steel. An assessment of the penetration of these high temperature gases into the upper head region will be pursued to follow up on this question. Modeling improvements could be related to adjustment of pressure loss parameters in the codes to reduce the degree of gas convection from the core to the upper head.

Question (to E. Raimond): Even with massive computing power, weather prediction is imprecise. Why do you believe that large-scale dispersion modeling will be successful?

Answer (E. Raimond): The objective is here to use data obtained from Meteo-France (the French organization in charge of weather forecast). Such large scale simulations are supposed to be useful for the post-accident management and to improve communication for public.

Chernobyl accident (all Europe was concerned by the accident) has clearly shown the interest for such simulations.

Weather prediction is supposed to be precise enough and would be periodically updated. Uncertainties are supposed to be more linked to the fission products transportation.

Question (to E. Raimond): Emergency preparedness in France assumes there will be 24 hrs before a serious release, correct? Are quick releases addressed considered realistic?

Answer (E. Raimond): In the operational context of a crisis, quick serious releases would be considered, in particular, in case of containment bypass and would be taken into account in the decision process for counter-measure.

Nevertheless, from the safety analysis point of view, it must be demonstrate that the probability of an accident with a serious release before 24 hours is very low.

The releases due to natural leakage of containment are of course taken into account from the beginning of accident.

Question (to E. Raimond): Could you describe in more detail the changes being made to source terms to take into account recent research?

Answer (E. Raimond): A reference source term (S3 - filtered release) has been updated in France for 1300 MWe and 900 MWe NPP by IRSN and for 900 MWe NPP by EDF. This update concerns mostly assumptions for iodine (gaseous form) and aerosol behavior in containment. This update is based on experimental results obtained before 2000.

Results of on-going International Source term Project will be used for quantification of uncertainties on source term. Main scenarios, from level 2 PSA, will be considered.

Question (to E. Raimond): What is the biggest impact on emergency planning expected from the ongoing work in France related to accident consequence analysis?

Answer (E. Raimond): On-going activities in France relative to accident consequence provide a better understanding of what could happen after a severe accident. In particular, level 2 PSA give quantitative information for a broad spectrum of scenarios. This knowledge will be used by the technical teams of crisis organization.

For NPP, the severe accident management guides are progressively improved in function of risk assessment with the objective to minimize the risk of a containment failure.

For the emergency preparedness, actions are on-going on the strategy of iodine prophylaxis and on long term post-accidental measures.

Question: What is your greatest fear with regard to the ability of the general public to understand or misunderstand the meaning of these analyses?

Answered: The planned SOARCA analyses are complex concepts developed through the use complex techniques. The challenge for the staff is to present the results in a clear and accurate manner that result in a common understanding for all stakeholders. Simply considering the number of potentially injured individuals for a particular event at a particular site can result in a significant misunderstanding of the result. The NRC will use risk communication techniques to best characterize the result (the consequences) in terms of risk. In addition, presenting multiple results to address the range of dose thresholds for a single event at a single site will also add to the difficulty for the public and other stakeholder to understand the results. To address this concern, the NRC is considering an expert elicitation to identify a single dose threshold distribution to present a single consequence for each event at each site.

Question: Has the NRC endorsed MAAP? If not, why and when? Will it and what can be done to get it endorsed?

Answer: The NRC has not performed a detailed technical review of MAAP and, therefore, has not endorsed MAAP. The industry / EPRI would have to make a formal request and be willing to pay the appropriate fees for the staff to review MAAP.

Question (to Farouk Eltawila (answered by Robert Prato): How does SOARCA affect license renewal, if any? Which pilot plants are selected? For those already received SER, For those in progress, For those to apply license renewal in the future?

Answer: SOARCA will not have a direct affect on license renewal. License renewal is a licensing action codified in Title 10 of the Code of Federal Regulations, Part 54 that is basically limited to the aging management of passive structures, systems and components. SOARCA is a research project that will be used to estimate the possible public health and safety consequences in the unlikely event of a commercial nuclear power plant accident releasing radioactive material into the environment. Because SOARCA is a research study, participation is voluntary. In addition, because of time and resources required to model plant design, conditions, and operations, the NRC is asking for volunteers that have the best models in place. License Renewal Applications completed, pending, or planned is not a criteria for considering a plant for the SOARCA project.

Question (to Farouk Eltawila (answered by Robert Prato**):** What plants will be studied? What is the schedule?

Answer: Currently, the following 5 plants have agreed to participate in the SOARCA project, Peach Bottom, Surry, Grand Gulf, Sequoyah, and LaSalle. Additional plants are being considered to complete the initial scoping criteria to include a representative from each of the different reactor-designs and containment-types in use in this country. We are currently planning on completing the first two studies by the end of the first quarter of 2008 and the remaining plants from the initial selection of plants by the end of 2009.

Question: Do you foresee or are there any plans to do an AST-2, RG 1.183 follow up and let industry audit SOARCA results for relief similar to NUREG-1465, etc?

Answer (R. Prato): Currently, RG 1.183 is tentatively scheduled to be revised in 2008. Industry will be given an opportunity to review and comment through public comment period for the draft for the proposed revision. No SOARCA specific information will be released until the project completion in late 2009. We will certainly consider the information developed from SOARCA as part of the revision process, but that may be limited at the time RG 1.183 is revised. Reg guides do not impose any requirements on licensees unless committed to by the licensee. If committed to a particular Reg guide, requesting relief from applicable requirements based on technical information is always an option.

Question: Do you expect significant benefit for a public consequence analysis if a design can show the reactor pressure maintains its integrity thought the accident sequence?

Answer: If the reactor coolant system pressure boundary is maintained throughout an event, there is very little, if any, potential for public consequences.

Question: What is the "probabilistic" element in level 3 PRA? Considering the amount of uncertainty involved in level 3 PRA, please clarify the "cost effectiveness" in performing level 3 PRA.

Answer: SOARCA is performing deterministic analyses to determine the potential consequences relating to public health and safety in the unlikely event of a radiological release to the environment. Deterministic modeling has advanced dramatically since 1982. Although, we recognize that uncertainty still exist in the result from these models, the primary objectives of SOARCA is to update the results from the 1982 Sandia Siting Study with the state-of-the-art methods and models, improved plant design and performance, and the knowledge gained from the past 25 years of national and international research.

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NRC 19th Annual Regulatory Information Conference
March 13-15, 2007
Bethesda North Marriott Conference Center

Session Title: GNEP and Fuel Cycle
Thursday, March 15, 2007

Question: Define TRU

Answer: TRU is defined as transuranic elements, also called transuranic actinides. These elements have an atomic number higher than the atomic number for uranium, or 92. The transuranic elements of greatest concern in GNEP are neptunium, plutonium, americium, and curium. TRUs comprise less than 1 percent of spent nuclear fuel, yet contribute the largest percentage of the long-term heat load and long-term dose to a potential geologic repository.

Question: The GNEP vision involves multiple passes through the irradiation-separation-fabrication cycle, and each pass would produce a slightly different version of the fuel. Would each version be separately qualified? If not, how will NRC account for the differences among the versions?

Answer (Giitter): Details about fuel qualification requirements will be developed after DOE determines specifically what type(s) of fuel will be fabricated at the CFTC, and how many passes through the ABR will occur. Once the details are determined, NRC will be involved both in fuel qualification and its impact on ABR safety. At that time, a single bounding fuel type or several fuel types may be qualified depending upon the fuel technical and safety performance from irradiation tests.

Question: You said it would be difficult for the ABR to meet the GDC on prompt negative reactivity feedback. Do you mean to suggest that NRC may drop this requirement for ABR licensing? If so, can this be justified on safety grounds?

Answer: Part 50, Appendix A, criterion 11, "Reactor Inherent Protection" (often called "GDC 11") requires the reactor core and associated cooling system to be designed so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity. This is often referred to as prompt negative reactivity feedback, or, simply, the increase in temperature causes a decrease in power without any control system activation. In existing water reactors, the water coolant has such a selfcompensating effect. However, with the fast neutron spectrum in sodium cooled reactors, the sodium coolant has a positive reactivity effect, and, if voids form from localized heating, the heating rate can continue to increase. Consequently, other features would have to be designed to compensate for this coolant effect and produce an overall, net negative reactivity feedback effect. Calculations have shown that fast reactor cores can be designed to utilize thermal expansion effects in the fuel assembly, fuel materials, and other core features (e.g., GEMs - gas expansion modules) to produce an overall net negative reactivity feedback. This has been shown in actual reactor testing during the 1970s and 1980s. Thus, while specific details would have to be reviewed in any sodium cooled reactor license application submitted to the NRC, it is likely this criterion can be met and the NRC would likely maintain the requirement in any future regulations.

The NRC would review any potential GNEP reactor application against other Appendix A criteria and apply them as appropriate to ensure safety. For example, the emergency core cooling system criterion may not be needed because the sodium coolant does not require pressurization to avoid boiling. On the other hand, regulation of a sodium cooled reactor would likely require criteria to address protection of the coolant from air and water. A new regulation with criteria applicable to sodium cooled reactors may be appropriate.

Question: What regulatory hurdles would be involved in storing and disposing of cesium and strontium separated out of spent nuclear fuel? Who would bear ultimate disposal responsibility?

Answer: The DOE is pursuing environmental impact statement (EIS) studies on GNEP and Greater-than-Class C (GTCC) LLW to identify and address the disposal of cesium and strontium. The NRC has not received any applications for disposal of GNEP cesium and strontium. However, if such an application is received, it would be reviewed using Part 61. GNEP wastes containing cesium and strontium would likely be GTCC LLW, with special handling, storage, and disposal requirements. Cesium and strontium wastes from GNEP might be in oxide or vitrified forms, in specially designed containers. DOE has indicated such waste forms might require storage for several centuries to allow the heat load and radioactivity to decay to allow for disposal as LLW. The NRC anticipates that the EIS studies and associated decisions will identify ultimate disposal responsibilities, likely commercial entities or governmental authorities.

Question: You said the uranium separated in the CFTC would most likely be disposed of as LLW. What is NRC's current position relative to the disposal of high U^{236} materials in near surface facilities?

Answer: Uranium recovered at the CFTC would be reprocessed uranium similar to that found in Europe and elsewhere. Such uranium contains around 0.4% of U^{236}, which is a neutron absorber. According to the current 10 CFR Part 61, a waste stream high in U^{236} would be considered Class A waste, which can be disposed of in a conventional near surface disposal facility. However, Part 61 was developed prior to the existence of new waste streams such as those that will be generated at the CFTC. As such, the NRC is currently evaluating the technical basis for Part 61 and its applicability to GNEP wastes. As a result of this evaluation, new guidance on disposal of uranium waste streams may be issued.

Question: GNEP has many international components and partners. Does the NRC foresee using MDEP (or an MDEP like process) to leverage international regulatory experience and knowledge in licensing GNEP facilities?

Answer: Depending on the level of engagement and interest of the international partners in GNEP, an MDEP like process may be beneficial for the GNEP program. The viability of and international interest in such standardized facilities will be gauged as the program progresses.

Question: CFTCs and ABRs would be subject to decommissioning financial assurance requirements, correct?

Answer: If these facilities are commercial facilities, then they would need to meet decommissioning financial assurance requirements.

Question: GNEP includes a small reactor component. What regulatory problems does NRC anticipate in regulating a small reactor program?

Answer (Pierson): At this point, because DOE has not developed the characteristics of the small reactor component of GNEP, it is too early to know what regulatory problems NRC may encounter.

Question: John Deutch of MIT called GNEP “goofy”. Can you explain why the industry and nuclear academics have not embraced Mr. Bush’s plan?

Answer (Killar): The industry supports the technologies that lead to closing the nuclear fuel cycle in a responsible way. These technologies, which under gird the GNEP program, are similar in many respects to the advanced fuel cycle initiative (AFCI) funded by Congress. However, GNEP is an international program that involves international negotiations. The industry will not participate in these discussions or in the international diplomacy that the administration will undertake to make the program a success.

Nuclear fuel recycling technologies and international "pooling" systems for fuel supply could enhance the nuclear non-proliferation regime and make the world safer. Advanced reprocessing and proliferation-resistant recycling technologies could recover 90 percent of the energy remaining in the fuel, guard against misuse of nuclear material, and reduce both the volume and radio toxicity of the remaining byproduct. However, the domestic nuclear industry is not a part of the international effort that encompasses the full sense of GNEP.

Question: The IAEA standard for self-protection is 100 rem/hr at 1 meter. You showed a dose rate for the pyroprocessing product of 200 rem/hr at 0.5 meter. For a point or line source, this would go below the IAEA standard at 1 meter, since the dose rate would be a factor of 2-4 times lower. Comment?

Answer (Saito): The issue of diversion would typically involve material in the kg scale. Therefore, a point or line source would be an oversimplification. The actual dose rate will be dependent on the age of fuel and burn-up from the light water reactor. The use of pyroprocessing will provide a final fuel form that has better detection capabilities than that of processes that produce purer forms of Pu. Being able to detect covert material theft makes diversion less likely.

Question: Why not use wet-type traditional reprocessing technology? New technology is risky, may fail, need much time and cost.

Answer: The pyroprocessing technology provides less risk than other options especially in regard to providing for a proliferation resistant process when compared to alternative proposals. The process has already been demonstrated with the EBRII fuel at INL and in many applications in the metallurgical (non-nuclear) industry. We believe that pyroprocessing will be proliferation resistant, have low environmental impact, is scaleable, and provides the most economic option.

The Committee on Electrometallurgical Techniques for DOE Spent Fuel Treatment was formed to evaluate the technical viability of electrometallurgical technology, the final report by National Research Council found in 2000 no technical barriers. This technology is ready to be commercialized.

<u>Frank Goldner</u>
Question: The planned throughput of the CFTC is 2,000-3,000 MTHM/yr, so 20-30 MT of Pu will be separated annually. How is this not sufficient to fabricate the first ABR core?

Question: Do you think it will be desirable to conduct reactivity insert accident experiments on transuranic fuels, such as those that have been conducted at labs with conventional fast reactor fuel?

Question: Could you please describe the likely impact of minor actinides on reactor safety parameters such as delayed neutron fraction and sodium void coefficient?

Question: Will multiple types of fuel require qualification in order to recycle the spent nuclear fuel multiple times? If so, how long might this take?

Question: In designing the AFCF, how can criticality safety and vessel shielding are accurately characterized, in light of the inherent uncertainties in source term development (e.g., burn up, enrichment, cross-sections) and discontinuities in pyrochemical and aqueous processes?

Buzz Savage
Question: Please discuss the uncertainty in meeting the 2020-2025 goals for having the GNEP facilities on-line and provide an estimate for meeting that goal.

Question: What are GNEP's resources?
FY 07 - authorized, expended/committed
FY 08 - President's Budget
FY 09 and Beyond - DOE requests and plan?

Question: Is a GNEP fuel cycle economically competitive with the existing once-through fuel cycle?

Question: Does DOE anticipate choosing one reprocessing technology and issuing an RFP for that particular method, or issuing a more general RFP and selecting one of several different methods proposed by industry?

Question: How do you intend to engage industry to obtain input regarding the June 2008 decision package?

Question: Is there a role for thermal recycling in GNEP and if not, why not (i.e., recycling in thermal reactors perhaps combined with recycling in fast reactors)?

Question: What difference, if any, is there between a "prototypical large-scale facility" and a "commercial-scale facility"?

Question: What effect would you expect of a change in administration on the future support and funding for GNEP by the US Government?

Question: What is the basis for not including the British in the partnership as they have substantial experience with the Sellafield site (same concept)?

Question: Connection between GNEP, including Russia involvement, and Putin initiative?

Question: In the business plan, what is the planned number of fast reactors, MW output?

Question: Why isn't there an option to ship existing LWR fuel overseas for reprocessing in GNEP? It would be a demonstration to the American public that this can be done safely plus it can be done now with an export license.

Question: Although recycling can increase the technical capacity of Yucca Mt, the legal 70,000 MT limit at Yucca Mt is only a 1982 political policy decision. The technical capacity of Yucca Mt is 300,000 MT, without recycling. The current legal 70,000 limit is not affected by GNEP, so why is the impression given that GNEP fixes Yucca Mt?

Question: Is it expected that commercial facilities implementing GNEP be built with appropriated funds?

Question: Since no utility will buy an ABR until the fuel has been qualified and proven, reactor safety has been proven, and economics have been confirmed, when will a fleet of ABRs become available?

Question: Please elaborate on your statement on "waste". Of the fuel cycle states you have identified which ones have committed or even raised the possibility of receiving and disposing of foreign nuclear waste?
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NRC 19th Annual Regulatory Information Conference
March 13-15, 2007
Bethesda North Marriott Conference Center

Session Title: Use of PRA Techniques for the Evaluation of Reactor Operating Experience

Thursday, March 15, 2007

Question: How do NRC event assessments model such things as common cause failures and component degradation?

Answer: As part of the Risk Assessment Standardization Project (RASP), procedures are being developed to provide standard guidance to analysts on how to handle these modeling issues in NRC risk assessments consistent with NUREG/CR-5485.

Question: How does the NRC plan on incorporating new reactor design and technologies into the SPAR Model Development Program?

Answer: The NRC has plans to incorporate new technologies (e.g., distributed control systems) that existing plants utilize in the plant-specific SPAR model on a case-by-case basis as these technologies come into use. The NRC also has plans to develop SPAR models for some of the advanced reactor designs. Both of these plans are still in the preliminary stages.

Question: What is the NRC's plan to incorporate Low Power and Shutdown (LP/SD), external events (e.g., fire, flooding, seismic), and Level 2 capability into SPAR models?

Answer: The NRC has developed some integrated SPAR models that include external events and LP/SD events. These models are being used in a trial basis for applicable risk assessments. The NRC is in the planning stages to provide quality assurance and validation for these models.

The NRC is in the early stages to develop the capability (for a subset of SPAR models) to probabilistically evaluate accident sequence progression from the core damage level through to containment failure, including an assessment of release magnitude and timing.

Question: How do we bring more convergence to plant PRA models and the SPAR models?

Answer: As part of the SPAR Model Development Program, all SPAR models are undergoing a process of performing a cutset level comparison and review with licensee's PRAs. Generic modeling issues are being worked on individual basis and a coordinated effort with industry is being planned.

Question: Is it possible that the NRC will use licensee PRAs in risk assessments in the future instead of the SPAR models?

Answer: A NRC/Industry working group has been formed and is reviewing this possibility.

Question: There tends to be misunderstandings between the licensee and the NRC on White findings (e.g., importance, assumptions, conservatisms, etc.)?

Answer: The NRC and licensees meet on each finding. The licensee is given time to comment and provide feedback on the NRC risk assessment. The NRC then reviews the comments to determine their validity and whether it affects the final color determination.
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NRC 19th Annual Regulatory Information Conference
March 13-15, 2007
Bethesda North Marriott Conference Center

Session Title: Emerging Issues: Electrical
Thursday, March 15, 2007

Chang-Fu Chuang, Section Chief
Taiwan Atomic Energy Council

Question: Please address environmental qualification program of your digital I&C system. Also address, the maintenance activities to prevent dust, heat, humidity of your installed I&C equipment.

Answer: The environmental qualification of our digital I&C system is based on IEEE Standard 323 and Regulatory Guide 1.89. Control of the electromagnetic environment follows the guidance contained in IEEE Standard 518 and EPRI TR-102323.

Question: How have you handled diversity in your I&C systems?

Answer: I&C equipment has a panel enclosure to prevent the intrusion of dust, heat, and humidity during shipment and storage. The I&C equipment is eventually installed in a HVAC environment. Generally speaking, there is an aluminum-type protection cover and a wood panel enclosure to protect the I&C Cabinets and equipment from humidity intrusion during shipment and storage. Upon receipt, the I&C equipment is stored in a Class A warehouse (temperature controlled between 18-23 degree centigrade, humidity below 50%). The wood panel enclosure and the aluminum-type protection cover are removed while performing the receiving inspection and maintenance activities in the warehouse. After inspection and maintenance, the aluminum-type protection cover is replaced to prevent dust and moisture from entering the I&C Cabinets and equipment. The I&C Cabinets and equipment are then moved from the warehouse to the field for installation provided that the field environment is in good condition. Once installed, HVAC is available to control the temperature and humidity.

Question: Do you model digital I&C systems in your probabilistic risk assessments (PRAs)? If so, how (that is, fault tree or dynamic modeling)? Do you use risk insights in your design? If so, how?

Answer: See response to next question.

Question: Do your designs include diverse digital or analog back-up systems? If so, what functions are backed-up by these systems? What regulatory guidance is there in your country regarding the need for diverse actuation systems?

Answer: Yes, The diverse I&C features are provided for protection against common mode failures of the protection systems. These features mitigate anticipated transient without scram (ATWS) events and ensure compliance with defense-in-depth requirements. Mitigation of common mode failures is provided by the following diverse features:

(1) Manual scram and main steam isolation valve isolation by the operator in the main control room in response to diverse parameter indications.

(2) Core makeup water capability from the diverse feedwater, control rod drive, and condensate systems.

(3) Availability of manual high pressure injection capability.

(4) Long term shutdown capability provided in a conventionally hardwired remote shutdown system (RSD) with 2 divisional panels containing analog or simple, dedicated and diverse software based digital equipment. Local displays of process variables in the RSD system are continuously powered and available for monitoring at any time.

The ATWS mitigation functions use diverse control logics from the primary protection system but are not necessarily hardwired:

(1) Alternate Rod Insertion, in association with the Rod Control and Information System.

(2) Fine Motion Control Rod Drive run-in

(3) Automatic Depressurization System inhibit

(4) Automatic Standby Liquid Control System initiation

(5) Feedwater Control System runback

NRC document SECY-93-087 and Standard Review Plan Section 7.8 are used as regulatory guidance.

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