RIC 2003 Questions/Take-aways

RIC 03 Panel Summary Form — Questions/Take-aways from W2 / NRC Chairman Nils J. Diaz

None.

RIC 03 Panel Summary Form — Questions/Take-aways from W3 / Safeguards & Security

Q: Need to clarify for licensees what they do when HSAS/NRC rating is at one level and state/local readiness level is different.
A: Licensees are advised by the NRC through a formal advisory process, normally following a determination by the Secretary of the Department of Homeland Security, to take action relative to HSAS protective measures developed by the NRC. At a minimum, the NRC anticipates that licensees will take action based on the Advisory they receive. Part of the protective measures in NRC guidance involves coordination with State and local law enforcement and emergency response organizations to allow for more informed decision making vis a vis deployment of limited resources. If the State or local government response level is different from the National Homeland Security Advisory System level, which may be the case, the licensee must evaluate the additional requirements, if any, and take any prudent action. It is important to note that HSAS advisories may relate to the nation as a whole, a particular geographic area or a specific sector such a nuclear power plants.

Additionally, NRC anticipates further dialogue on this issue in an Outreach Session sponsored jointly by the NRC and DHS to be conducted for State Homeland Security Advisors, Radiation Control Officers, and State Liaison Officers scheduled at NRC Headquarters in June 2003.

Q: *Tom Lockwood* - provide an overview on the role of the states' national guard assets in deterring or responding to an attack and any mechanism (planned or in place) for meeting the goal(s). (From J. Tomlinson)
A: In response to the events of September 11, 2001, following dialogue with the NRC Chairman, many State Governors decided to deploy National Guard assets to protect nuclear power plants and other critical infrastructure components within their States. Governors have the authority and responsibility to deploy these and other State assets based on their understanding of the threat, determination of appropriate action, and available resources to respond to it. The NRC has stressed coordination among our licensees and state and local authorities in our guidance reflected in NRC's protective measures for the Homeland Security Advisory System threat levels. NRC will continue to work with States to ensure that nuclear power plants remain a well protected element of the nation's critical infrastructure.

Following questions received in Safety Culture (W8) session but referred to W3 for reply...

Q: Regarding Public Confidence/NRC's credibility: How bad is a worst case?
A: The mission of the NRC is to protect the public health and the environment. In carrying out our safety mission, the NRC is committed to ensuring that there are no acute radiation exposures resulting in fatalities and no releases of radioactive materials that result in significant radiation exposures. The occurrence of either of these is considered unacceptable. These goals are captured in the NRC strategic plan and drive our resources, which are dedicated to protecting the public health and environment. In fact, the NRC has a full range of actions at its disposal to help ensure that any licensee's performance does not fall below acceptable levels. In the rare cases when licensee performance has been determined to be unacceptable, the NRC has taken actions it deemed appropriate to protect the public's health and the environment.

Estimates of reactor accident consequences have evolved over the years along with our improved capability for analysis. Our extensive experience and analysis shows that the probability of a severe accident is very low. Even events that have resulted from equipment failures, degraded conditions, or human errors (or a combination of all three) have never resulted in a significant release of radiation that would affect public health.

Determining the potential extent of a radiological release or public impact requires a plant-specific quantitative analysis that is based on the likelihood that a combination of plant equipment or human failures would occur along with a determination of the associated consequences. This analysis is called a probabilistic risk assessment and is the best source of information to respond to this question. Such analyses have been performed at virtually every operating reactor in the U.S. The results have shown that, even if a significant plant initiating event were to occur, the various barriers, plant features and emergency plans in place would keep the impact on the public relatively small.

RIC 03 Panel Summary Form	Questions/Take-aways from W3 / Safeguards & Security

Q: Are dangers properly ID'd?
A: Although there have been no specific credible threats of a terrorist attack on nuclear power plants since September 11, 2001, the NRC works closely with the intelligence and law enforcement community, including the Department of Homeland Security and others, to assess any threats that may be directed towards nuclear facilities. Although it is difficult to predict when and where terrorists may strike next, the robust security at nuclear plants should serve as a significant deterrent. It should be recognized that nuclear power plants are massive structures with thick exterior walls and interior barriers of reinforced concrete. The plants are designed to withstand tornadoes, hurricanes, fires, floods, and earthquakes. As a result, the structures inherently afford a measure of protection against any form of attack. In addition, the defense-in-depth philosophy used in nuclear facility design means that plants have redundant and separated systems in order to ensure safety. That is, active components, such as pumps, have backups as part of the basic design philosophy. This provides a capability to respond to a variety of events.

It is also important to note that nuclear power plants have a robust emergency preparedness program that includes biennial, evaluated exercises. In the event of a serious problem including a terrorist attack around a nuclear power plant, the plans and procedures that have been routinely exercised would be activated. This provides a capability to respond to events of all types.

In addition, since September 11, 2001, the NRC has undertaken a systematic process to identify potential vulnerabilities to various aspects of NRC licensed facilities and activities. As a result, we have issued a number of security advisories and orders to various categories of NRC licensees to provide guidance and requirements on actions necessary to protect their activities and facilities in the new threat environment.

Q: If successful terror act unacceptable [sic], is phase out of [nuclear power plants] an option?
A: The NRC does not consider the risk from a potential terrorist attack to be unacceptable at NRC licensed facilities. We have taken significant action including issuing advisories and orders to enhance security at those facilities. However, NRC retains the option of ordering nuclear power plants to cease operations if we determine that continued operations are no longer being conducted in the interest of public health and safety.

RIC 03 Panel Summary Form	Questions/Take-aways from W4 / Stakeholder Forum

Q: Consider adding Public Confidence & EP to next year's RIC topics.
A: We have already begun planning for the 2004 RIC agenda and will add these topics to our list of possible topics.

RIC 03 Panel Summary Form	Questions/Take-aways from W5 / Reactor Oversight Process

Q: The ROP specifically calls for no aggregation of findings to increase an issue's significance. How does the practice of calling out a "substantive cross-cutting issue" in the annual assessment letters match up with this guidance?
A: While it is true that the ROP does not provide for the aggregation of green inspection findings into a finding of more significance (i.e., white), the ROP does allow for the identification of a substantive cross-cutting issue in one of the three previously defined cross-cutting areas (human performance, problem identification and resolution, safety conscious work environment). The criteria for identifying a substantive cross-cutting issue are contained in Inspection Manual Chapter 0305. The process of identifying a substantive cross-cutting issue is meant to provide early warning to licensees about what is perceived to be a potential performance problem in a cross-cutting area, even before the performance problem results in an actual issue of risk significance. No additional supplemental inspections are conducted as a result of identifying the cross-cutting issue.

RIC 03 Panel Summary Form	Questions/Take-aways from W6 / Steam Generator & Materials Issues

None.

RIC 03 Panel Summary Form	Questions/Take-aways from W7 / License Renewal/Power Uprate

None.

RIC 03 Panel Summary Form	Questions/Take-aways from W8 / Safety Management

NRC & INPO

Q: Everyone agrees that the Davis Besse event implies that somehow the licensee failed, the NRC failed, and INPO failed. What will the NRC & INPO be doing differently in the future to ensure this never happens again?
A: After the degradation of the reactor vessel head at Davis-Besse was realized, the NRC formed a Lessons Learned Task Force to review the full scope of NRC regulatory activities related to the incident, including the agency inspection and assessment program, industry-wide generic activities, research work, and international practices. The Task Force concluded that the reactor head damage was preventable, but was not prevented because: (1) the NRC, Davis-Besse, and the nuclear industry failed to adequately review, assess, and follow up on relevant operating experience; (2) Davis-Besse failed to assure that plant safety issues would receive appropriate attention: and (3) the NRC failed to integrate known or available information into its assessments of Davis-Besse's safety performance.

The Task Force's findings and recommendations were reviewed by a panel of senior NRC managers and four action plans were developed to address the findings. These plans address:

- Assessment of Stress Corrosion Cracking: This activity involves the review of plant experience with stress corrosion cracking and boric acid corrosion in order to provide a basis for revising inspection requirements and developing improved inspection program guidelines.

- Assessment of Operating Experience, Integration of Operating Experience Into Training, and Review of Program Effectiveness: This activity includes programmatic improvements for review of operating experience and enhancements to inspector training and qualification.

- Evaluation of Inspection, Assessment, and Project Management Guidance: This activity contains three distinct topics: (1) changes to the NRC's inspection program to ensure that long-standing unresolved problems receive sufficient inspection (2) development of guidance to assess the impacts of Inspection Manual Chapter 0350 on regional resource allocations, and (3) development of guidance to ensure that decisions to allow deviations from agency guidelines in generic communications are adequately documented.

- Assessment of Barrier Integrity Requirements: This activity includes improvements to reactor coolant leakage detection requirements and performance indicators.

L. Jarriel

Q: In your understanding of safety culture, is technical competence a factor? If so, how important is it?
A: Technical competence plays a role in safety culture in that it contributes to achieving many of the attributes that define a strong safety culture; such as a questioning attitude, a rigorous approach to operations and corrective actions, an effective learning environment, and appropriate prioritization of issues requiring resolution.

RIC 03 Panel Summary Form	Questions/Take-aways from W8 / Safety Management

L. Williams

Q: Where are the 23 indicators you mentioned published?
A: In the presentation by Laurence Williams, Her Majesty's Chief Inspector of Nuclear Installations and Director of the United Kingdom's Health & Safety Executive's (HSE) Nuclear Safety Directorate, 23 indicators of a positive safety culture were mentioned. These indicators, as defined by HSE, are:

1. Visible leadership and commitment of top management
2. Organisational value systems
3. Safety role of line management
4. Strategic business importance of safety
5. Supportive organisational culture
6. Involvement of employees
7. Organisational learning
8. Measurement of safety performance
9. Mutual trust and confidence between management and workforce
10. Effective communications
11. Absence of safety versus production conflict
12. Demonstration of care for all those affected by the business
13. Organisational support
14. Openness
15. Timely processing of corrective actions
16. Quality of problem and change analysis
17. Independent review processes
18. Violations monitoring and reduction
19. Monitoring of requests for special dispensation
20. Monitoring of excessive hours of work
21. Assessment of adequacy of training
22. Use of suitably qualified and experienced persons
23. Understanding of Job descriptions

RIC 03 Panel Summary Form	Questions/Take-aways from W8 / Safety Management

L. Jarriel

Q: Given that safety culture was a root cause of a significant incident at Davis-Besse, how can the public have confidence in the NRC if it does not engage in safety culture regulation?
A: The NRC does not have specific regulations that directly relate to safety culture. However, the NRC seeks to ensure the existence of an appropriate safety culture through a variety of indirect means.

In 1989, the NRC published a policy statement entitled "Conduct of Nuclear Power Plant Operations." It stated that "management has a duty and obligation to foster the development of a 'safety culture' at each facility..." In defining safety culture, the NRC referenced the International Safety Advisory Group definition.

A premise of the NRC's Reactor Oversight Process is that a licensee's poor safety culture will become evident through various performance indicators and safety findings. Weaknesses in the licensees' operations will emerge during baseline inspection activities. The baseline inspection program looks at licensees' processes that provide important insights into its safety culture such as problem identification and corrective action. Verification of licensees' implementation of the maintenance rule and the inservice inspection program, which are also part of the baseline inspection program, provides assurance that a sensitivity to risk-significant activities is part of the normal operating framework. Also, the NRC's examination of allegations provides insight into the willingness of a licensee to receive and address employee concerns. Thus, important aspects of safety culture are routinely evaluated at operating plants. In the case of Davis-Besse, the NRC recognizes that there were opportunities to identify the declining safety culture at the site before the reactor vessel head degraded condition was found. The Agency has taken steps to address potential deficiencies in the Reactor Oversight Program's baseline inspection and performance assessment processes in order to improve the ability to identify declining trends earlier.

As the NRC assesses the impacts of weak safety culture on plant operations, it will continue to work with the international community to better define the concept of safety culture and determine additional roles the regulator should play.

RIC 03 Panel Summary Form	Questions/Take-aways from T1 / Building on the Davis Besse Experience

Q: UCS's review of Davis Besse indicated that the Boric Acid Corrosion Generic Letter had not been incorporated into the design basis. FirstEnergy signed the 50/54 (f) letter that the design basis would be maintained. Why has NRC failed to take enforcement action?
A: The licensee developed an implementing procedure in response to the Boric Acid Corrosion Generic Letter (GL 88-05); thus, activities discussed in GL 88-05 were incorporated into the Davis-Besse design basis. The procedure is referenced in the NRC Augmented Inspection Team inspection report dated May 3, 2002 (ADAMS accession no. ML021260141), which contains the following statement: "Davis-Besse's implementing procedure for GL 88-05 was NG-EN-00324, 'Boric Acid Corrosion Control.'"

The Davis-Besse licensee responded to our letter dated October 9, 1996, concerning adequacy and availability of design bases information, by letters dated February 11, and March 31, 1997, and December 17, 1999. Also, by letter dated November 20, 2003, the licensee submitted additional design basis information to the NRC, which includes consideration of their recent system design reviews.

A final Red finding was issued to Davis-Besse on May 29, 2003, for the performance deficiency associated with failure to properly implement the boric acid corrosion control program and corrective action program. As specified in that letter, no Notice of Violation was attached at that time due to ongoing criminal investigations into the cause of the violations. Once these investigations are complete, the NRC will consider enforcement actions as appropriate.

Q: NRC response appears to be too narrowly focused upon the vessel head & missed the programmatic problems that led to the "enormous failure" in NRC regulation. Please address.
A: The Davis-Besse Lessons Learned Task Force (LLTF) report, dated September 30, 2002, and publicly available through ADAMS package accession number ML022740211, was an evaluation of the NRC's regulatory processes related to assuring reactor vessel head integrity. It also included recommendations for improvement for both the NRC and the industry in a number of regulatory program areas. As a result, the NRC has developed Task Action Plans to internally address four key programmatic areas for improvement: 1) the assessment of stress corrosion cracking, 2) the use of industry operating experience and generic communications, 3) managing regional resource allocations and inspection programs, and 4) strengthening barrier integrity requirements.

Q: The LLTF was "independent," but was essentially comprised of NRC staff. What value would be added by having regular independent assessment by a team of all stakeholders to assess the effectiveness of the ROP, public access to information and public confidence in the NRC?
A: An independent assessment by a team representing a wide variety of stakeholders was conducted to evaluate the effectiveness of the ROP pilot program in 2000 and another independent assessment was conducted to evaluate the ROP's first year of initial implementation in 2001. The results of these assessments are available on the NRC's web site at http://www.nrc.gov/reactors/operating/oversight/pilot-program.html and http://www.nrc.gov/reactors/operating/oversight/initial-evaluation-panel.html, respectively. These teams were established by the agency in accordance with the requirements of the Federal Advisory Committees Act, and were very resource intensive for both the staff and the stakeholders.

The NRC staff currently conducts a self-assessment each year using inputs from all stakeholders. We actively seek this input and receive feedback on all aspects of the ROP, including public access to information. In addition, the staff receives feedback on a continuous basis through its web site, written correspondence, public meetings with utilities, conferences, public meetings with industry representatives, and Commission meetings. Finally, the NRC's Advisory Committee on Reactor Safety (ACRS) provides independent reviews of various aspects of the ROP on an ongoing basis. The most recent staff self-assessment is in "SECY-03-0062, "Reactor Oversight Process Self-assessment Program CY 2002," available at http://www.nrc.gov/reading-rm/doc-collections/commission/secys/. Previous self-assessments are in SECY-02-0062 and SECY-01-0114, available on the same web site.

The NRC periodically conducts stakeholder workshops to discuss public involvement issues. Input received during these workshops informs changes to policies on public involvement and public confidence issues. For example, the revised policy on public meetings was significantly enhanced by input received at a stakeholder workshop. In the next several months, the NRC plans to conduct another stakeholder workshop. The workshop will be announced on the NRC's web site, http://www.nrc.gov/public-involve/public-meetings/meeting-schedule.html.

Q: Failure to promote a safety conscious work environment (SCWE had a role in the Davis-Besse situation. Is the NRC planning to revise their inspection process? What are the planned revisions?
A: The Commission recently directed the staff to develop guidance, in consultation with stakeholders, for licensees and inspectors concerning best practices to encourage a SCWE. It is not known at this time what format such guidance will take, but the process for developing it will be public. In addition, the Davis-Besse Lessons Learned Task Force recommended that the staff examine its oversight of this area. As a result, the staff intends to review the adequacy of NRC baseline inspections and plant assessment processes. Information on the Davis-Besse vessel head degradation, including the LLTF report and the staff's actions in response, can be found at: http://www.nrc.gov/reactors/operating/ops-experience/vessel-head-degradation.html.

RIC 03 Panel Summary Form	Questions/Take-aways from T1 / Building on the Davis Besse Experience

Q: *For FENOC VP* - The BWOG undertook a safety and performance improvement program SPIP in the mid 80's. What if anything has FENOC learned from the review of that program today?
A: The BWOG effort of the mid-80's was not recently specifically reviewed by FENOC. At the time, it provided collective operating experience of B&W plants. It dealt primarily with transients and scram reduction efforts. Recognizing it was over 15 years ago, the program has given way to a much more thorough and mature industry operating experience approach.

Q: *For FENOC VP* - Who presents in the Davis Besse Board oversite meetings, the line organization or independent groupes such as QA, onsite and offsite safety committees, etc.?
A: Both groups (i.e., line management and oversight) attend, routinely present, and participate in the discussions at all levels. The Nuclear Committee and the Oversight Committees have the option to meet separately with the oversight executive or the line management executive.

Q: *For FENOC VP* - Many of us have assessed safety culture at sometime before the D/B "event." Assuming that D/B did also, what was learned from these previous assessments and how will this be applied to assessments going forward?
A: Previous assessments of selected aspects of safety culture were performed at Davis-Besse, by both internal and external organizations. These assessments identified some specific attributes for improvement in areas such as engineering and corrective action. Based upon our Davis-Besse experience, we have strengthened these assessments to be more comprehensive and to ensure improved follow-up on identified issues.

Q: *For FENOC VP* - Can you describe specific actions implemented on containment sump? What is the relationship of these actions and Davis Besse RPV head degradation?
A: The containment sump has been modified to add substantial strainer capacity. While not specifically related to the RPV head, the sump was one of several modifications to the plant to add design margin.

Q: It is clear that Region III does not agree w/ many of the conclusions in the "lessons Learned" report. How will NRC alignment and "buy in" in actions going forward be ensure, particularly in Region III? (Deferred to F2?)
A: This question was answered directly by Mr. Jim Dyer, Region III Administrator, during the Davis Besse and the Region III breakout session.

RIC 03 Panel Summary Form	Questions/Take-aways from T1 / Building on the Davis Besse Experience

Q: Lessons Learned Task Force indicated that PRAs were insufficient and ROP blinded First Energy and NRC because the PRAs don't recognize that the RPV could fail. Both of these problems were identified in the ROP rollout years ago. Focusing on your inspectors will not solve the glaring problems in the ROP and the PRA blinders placed on the staff. Will NRC require new PRAs that reflect the reality of RPV failure?
A: The NRC does not require licensees to evaluate Reactor Pressure Vessel (RPV) failure or any other accidents that are addressed in current PRAs. There are no regulatory requirements for licensees to evaluate the probabilistic risk of plant operation. However, in accordance with GL 88-20, the NRC requested that licensees submit an Individual Plant Examination (IPE) for internal plant hazards (e.g., LOCAs, loss of offsite power, general transients, etc.) and an IPE for External Events (IPEEE). External event hazards primarily include fires, floods, and high wind events. The purpose of GL 88-20 and its supplements was to identify any plant vulnerabilities and when deemed prudent by licensees that voluntary actions be taken to reduce the risk of these vulnerabilities.

When the IPEs were being developed, the risk of RPV failure was considered. However, the estimated frequency of an RPV failure per year of plant operation is very low. Because this initiating event frequency is very low, most PRAs did not further evaluate this event into the overall core damage frequency calculation. Even if this initiating event had been evaluated, the PRA results would not have been probabilistically important towards identifying plant vulnerabilities. The PRAs did however evaluate the risk of various LOCA events. Three LOCA sizes were evaluated which are the small, medium, and large LOCA events.

The Reactor Oversight Process (ROP) uses the Significance Determination Process (SDP) to determine the significance of licensee performance deficiencies. The SDP attempts to determine the change in core damage frequency resulting from these deficiencies. For plant operational events, the conditional core damage probability is evaluated. Both these processes use risk tools, including information from licensee PRAs to determine risk significance and an appropriate regulatory response. These processes use the "best available information" to determine the overall risk effects. Regarding RPV failure, the best available information indicates that RPV failures continue to represent a very low risk to plant operation because the RPV is very robust. However, in the case where a licensee does not adequately implement required programs such as a corrective action program, the potential does exist that the robustness of the RPV could be potentially compromised. When new information is discovered that is risk significant, the NRC reacts aggressively to better understand the impact of that new information. This new information then becomes part of the best available information discussed above and is factored into the overall NRC regulatory response, not only for the associated licensee, but generically for the industry.

It is important to recognize that the ROP is a comprehensive evaluation of licensee performance both through inspection findings and performance indicators. Inspection procedures and performance indicators are based on years of NRC experience regulating the nuclear industry. Although PRA is an important element of the ROP in defining risk significance, it does not stand alone as the NRC's sole source of information for understanding licensee performance. In particular, the NRC's assessment process includes activities that evaluate a licensee's ability to identify and correct problems. This aspect of the ROP forms a foundation for the overall understanding of licensee performance. If a licensee has been determined to be ineffective or incapable of correcting their problems, serious regulatory action can be taken to ensure public health and safety is maintained.

RIC 03 Panel Summary Form	Questions/Take-aways from T2 / Emerging Technical Issues

None.

RIC 03 Panel Summary Form	Questions/Take-aways from T3 / Risk Informed Activities

Q: What are the future implications of the increased emphasis on “risk-informed regulation”? Does this imply complementary use of deterministic and probabilistic approaches in safety cases?
A: The staff will continue to attempt to implement the Commission PRA Policy Statement and move to a more risk-informed and performance-based regulatory structure. It should be noted that moving to this structure is predicated on the industry’s performance being maintained at high levels. As the state-of-the-art in PRA methods and data improves, the staff will look for additional opportunities to use PRA to enhance safety, reduce unnecessary regulatory burden, and make our processes more effective, efficient, and realistic. A risk-informed and performance-based regulatory structure inherently balances deterministic and probabilistic approaches in safety cases.

RIC 03 Panel Summary Form	Questions/Take-aways from T3 / Risk Informed Activities

Q: Has the periodic re-evaluation of risk significance of structures, systems, and components (SSCs) determined that a low or non-risk significant SSC should be considered high risk significant? If so, was it based on performance degradation due to reduced treatment?
A: There have been a few instances where a previously categorized Low or NRS SSC has had it's categorization raised to a safety significant level (Medium or High). South Texas Project (STP) views these changes as 'critical changes,' and places a higher emphasis on understanding the basis for these changes. STP incorporates procedural processes to reinstate special treatment requirements on these safety significant components in a managed fashion once the change basis is communicated and documented.

Revising a component categorization from a lower level to a higher level can be due to a number of reasons. New insights into industry operating experience has been a basis to retain special treatment requirements on components whose performance requires heightened attention and monitoring. In addition, revisions to PRA modeling can also affect component categorizations. Generic changes to PRA models can result in component categorization adjustments, and these changes should be thoroughly understood prior to implementation.

In STP's experience to date, the periodic review process of component categorization has resulted in no component categorizations being raised due to performance degradation resulting from reduced treatment.

Q: For those SSCs determined to be non-safety significant, is there a change introduced in the PRA in terms of increased failure probabilities for these components due to commercial procurement?
A: There is no specific change introduced into the PRA for potential increased failure probabilities due to applications of industrial replacement components. STP firmly believes that since the design requirements are not changed for the low safety significant components, purchasing a replacement commercial component from a reputable vendor will result in no decrease in performance for the affected SSC. In fact, reviews of historical industry data have shown no significant performance differences between safety-related and industrial components for a wide variety of component types.

However, STP does include a sensitivity study in the PRA to evaluate the impact of a postulated ten times increase in failure likelihood for all modeled Low risk ranked SSCs. While STP does not expect the performance of Low safety significant components to be affected by reduced treatment, this sensitivity study was viewed as a bounding case for consideration. The sensitivity study showed that if all modeled Low ranked components' failure rates increased simultaneously by a factor of ten, the resulting Core Damage Frequency (CDF) increased by less than 1%, while the resulting Large Early Release Frequency (LERF) increased by less than 0.5%.

By performing periodic reviews and through continuing system engineering evaluations of system and component performance, it is expected that any component performance degradation will be noted and acted upon well before the failure rate approaches a factor of ten increase.

Q: Switching from "safety related" to "safety significant" classification reveals many safety related components have low safety significance. How has STP managed the change in culture necessary to make this transition- especially since it may indicate that tasks that were thought important for 30 years are no longer important?
A: A couple of points are important to note: safety related components will remain safety related with the implementation of Option 2. The component classification remains unchanged; rather, the component has been categorized. With this new insight, the component's overall safety significance is better understood. Secondly, the workforce generally understands what makes sense in the operation and maintenance of the equipment. NRC regulations require that certain types of tests or inspections, and their associated frequencies, be performed for all applicable safety-related components without consideration of the component's relative safety significance.

Option 2 permits a documented basis for categorizing components as either safety significant or as non-safety significant. The safety significant components will retain the treatment requirements that the regulations have dictated for years. Non-safety significant components, while still safety-related, are permitted to have reduced treatment (a commercial style of approach) applied to them. Having this importance basis documented and applying the right type of treatment commensurate with the component's importance inherently makes sense to the industry. While there may be reluctance to make this change (due to the instilled nuclear mind-set of the past 30 years), increasing the communication on these milestone changes and searching out key change-agents in your organization to help foster the transition will aid in getting the general workforce on board. Our general experience is that the workforce is more accepting of these changes than some of the management and engineering teams.

RIC 03 Panel Summary Form	Questions/Take-aways from T3 / Risk Informed Activities

Q: What are the key cultural pre-requisites to a successful implementation of Option 2?
A: The key cultural pre-requisite is to have a strong, established safety culture already in place. A common misconception is that Option 2 merely allows reduced treatment for low safety significant components. While this is a benefit of Option 2, the primary mission of Option 2 is to increase overall nuclear safety.

A successful Option 2 implementation leads the organization into a greater awareness of nuclear safety and the components that support it. This allows for a proper allocation of resources, planning, preparation, oversight, etc. for activities that affect safety significant components. For components that are safety significant, the organization should readily agree that performing the full scope of obligated tests, inspections, and maintenance is necessary to ensure safe, reliable performance. These activities should be approached with a recognized sense of the role these components play in contributing to nuclear safety.

On the other hand, activities performed on low or non-safety significant components are not relegated to poor performance or sloppy workmanship. It is still expected that these components will reliably perform their function in an efficient manner.

Lastly, it is important to recognize that the categorization process may identify a small number of non-safety related components that are safety significant. The organization should accept the possibility of increased controls on these components, consistent with the goal of maintaining or enhancing overall nuclear safety.

Without a strong safety culture currently in place, it would be difficult to reinforce the foundational principles of a successful Option 2 without reverting back to an approach that simply views Option 2 as another way of saving money.

RIC 03 Panel Summary Form	Questions/Take-aways from T4 / “Spent Fuel Storage, Transportation, and Disposal”

Panelist Christopher Kouts, U.S. Department of Energy:

Q: You did not mention quality assurance (QA) as a challenge. Is it not a program area challenge?
A: Quality Assurance will be an important factor in the Yucca Mountain licensing proceedings. For this reason the Director of the Office of Civilian Radioactive Waste Management has made the refinement and implementation of Quality Assurance procedures one of the program’s highest priorities. This is reflected in the Management Improvement Initiative, issued by Dr. Margaret Chu, that highlights Quality Assurance in three of the program’s five key initiatives.

Q: Has DOE decided on the "hot versus cold" design for the repository? If not, when will the decision be made?
A: The Total System Performance Assessment (TSPA) in the license application will be based on a higher temperature operating mode. However, the repository design and concept of operations will provide the flexibility to close the repository in either "hot" or "cold" thermal modes.

Q: The NRC has an extensive process in place to license spent nuclear fuel transport casks. When will DOE acknowledge the acceptability of these NRC licensed transport casks for transport of spent nuclear fuel to DOE?
A: The Nuclear Waste Policy Act states that no spent nuclear fuel or high-level radioactive waste may be transported by or for the Secretary under the Nuclear Waste Policy Act except in packages that have been certified for such purpose by the Commission. The Department has always acknowledged the acceptability of these NRC licensed transport casks for transport of spent nuclear fuel to DOE.

Q: Has DOE finalized the design of the disposal canisters to accommodate dual purpose canisters such that these canisters do not have to be reopened to be placed in the disposalcanister?
A: DOE has not yet finalized the design of the disposal canisters for the repository (also referred to as waste packages); however, current design plans do not anticipate that dual purposes canisters will be loaded directly into the waste package.

Q: When will DOE select licensing counsel for the Yucca Mountain licensing proceeding?
A: The Department recognizes increased legal resources will be needed during the licensing process and has been analyzing how to best secure such resources. We expect to announce our approach on securing those resources in the near future.

Q: Given that site recommendation is complete, isn’t it time for DOE and the nuclear industry to settle the lawsuit over untimely fuel receipt?
A: Currently 23 utilities have filed 21 pending lawsuits in the Court of Federal Claims for breach of contract. The contract breaches are related to the Department's delay beyond 1998 in accepting spent nuclear fuel from utility contract holders and are not linked to the recommendation of the site. Settlement of issues in litigation is always an option available to the parties; however, the Department does not comment on matters in pending litigation.

Q: Who at DOE is responsible for Greater than Class C (GTCC) waste planning and disposal? Who does DOE believe is their counterpart at NRC?
A: According to the Nuclear Waste Policy Act, the DOE Office of Civilian Radioactive Waste Management is responsible for the disposal of spent nuclear fuel and high-level radioactive waste. The DOE Office of Environmental Management is responsible for GTCC low-level waste currently managed by DOE. DOE is also currently planning the initiation of an Environmental Impact Statement (EIS) process to analyze the alternatives for disposal of GTCC low-level waste.

We believe that the responsible regulatory office at the NRC for GTCC low-level waste is the Office of Nuclear Materials, Safeguards and Security.

Q: What person, committee or department within DOE is responsible for innovative thinking or finding new thinking on spent fuel management? Do you agree advance management concepts must be considered, beyond a site in Nevada?
A: The Office of Civilian Radioactive Waste Management is directed by the Nuclear Waste Policy Act to conduct its mission to develop an NRC-licensed repository at Yucca Mountain. Alternative technologies have been, and will continue to be evaluated for the responsible management of high-level radioactive waste. For example, the Department supports and continues to fund further research and development of accelerator transmutation of nuclear waste. The DOE offices responsible for innovative and new thinking on spent fuel management, not related to the development of an NRC-licensed repository, include the Office of Nuclear Energy, Science and Technology, the Office of Policy and International Affairs, the Office of the Under Secretary, and the Office of Environmental Management.

RIC 03 Panel Summary Form	Questions/Take-aways from T4 / "Spent Fuel Storage, Transportation, and Disposal"

Panelist Bill Brach, U.S. NRC:

Q: What is going on at Sandia who is looking at transport cask sabotage risks (e.g., dollars planned, scheduled work, report schedule)?
A: NRC has a number of activities underway to examine transport cask vulnerability to terrorist activities. Sandia is providing significant technical support to NRC in this review. Much of this work was initiated after the September 11, 2001 terrorists events. The reviews and contract support activities are currently underway. We hope to have results from this study available by end of the year. Preliminary information to date has confirmed the robustness of these casks to withstand significant physical challenges of terrorists events. Upon completion of the studies, NRC will determine if any additional cask design requirements, physical protection requirements, or administrative control requirements may be needed. The funding for these studies was provided by non-fee base funding from Congress.

Q: Given the availability of the general license for independent spent fuel storage installations (ISFSI's) at licensed facilities without operating reactors, why should licensees seek a site-specific license?
A: The option is available to licensees to select either a general or specific license path for their ISFSI. The majority of licensees have selected the ISFSI general license, while a few have selected the ISFSI specific license. There is not a right or best path, rather the option is left to the licensee to select the path based on their own identified needs.

Q: Given limited resources, and pending risk-informed improvements in LOCA, should NRC be streamlining their regulations in Part 71 and Part 72 relative to changes?
A: The reference to pending risk informed improvements in LOCA (loss of coolant accident as used in the Part 50 power reactor arena) is not clear in relation to regulation of transportation under Part 71 or storage under Part 72. However, the NRC strives to incorporate risk information and performance based considerations in all of our regulated activities, including Parts 71 and 72.

Q: Where does NRC stand on credit for spent fuel burn-up to allow more fuel assemblies to be loaded in shipping casks?
A: NRC's Spent Fuel Project Office issued a new interim staff guidance document on use of burnup credit in both transportation and storage in the fall of 2002. The new position expands the staff's position to allow credit for actinides and other considerations. The ISG is available publicly and can be accessed on the NRC web page at the following URL address: www.NRC/GOV/reading-rm/doc-collections/isg/spent-fuel.html

Q: At decommissioned plants today, spent fuel is stored for shipment. Who at NRC is responsible for the new need for careful documentation and software legacy (that is, to assure licensees manage documentation carefully) in preparation for transportation/shipping to a future center of management or disposal?
A: Parts 71 and 72 identify the records and management systems and quality assurance requirements necessary to support transportation and storage of spent fuel. The licensee is responsible for the generation and maintenance of these record and management systems, and full compliance with the quality assurance requirements. Through periodic NRC inspection, NRC reviews the adequacy of the licensee's actions to comply with these requirements.

Q: Approximately how many storage casks have been loaded and are in use at NRC licensed sites?
A: This question was partially answered at the RIC. The answer provided by the NRC was that there are approximately 200- 400 casks loaded with spent fuel at NRC licensed ISFSIs. After the RIC session, an industry representative indicated that to his knowledge there are about 380 loaded casks, which is within the range provided by the NRC during the session.

Q: What person, committee or department within NRC is responsible for innovative thinking or finding new thinking on spent fuel management? Do you agree advance management concepts must be considered, beyond a site in Nevada?
A: Congress established NRC to focus on the licensing and regulatory functions, all other activities remained with what is now the Department of Energy. NRC mission is to regulate the Nation's civilian use of byproduct, source and special nuclear material to ensure adequate protection of public Health and safety, to promote the common defense and security, and to protect the environment. The Spent Fuel Project Office has the licensing and related regulatory functions for the safe storage of spent nuclear fuel in a independent spent fuel storage installation. NRC's role is to assure that nuclear technology can be used and used safely, therefore NRC focus is to see that optimization is done safely.

RIC 03 Panel Summary Form	Questions/Take-aways from T4 / “Spent Fuel Storage, Transportation, and Disposal”

Panelist Janet Schlueter, U.S. NRC:

Q: NRC has never licensed a geologic repository. What steps are being taken today to ensure that NRC’s repository licensing team will be properly staffed and funded with a stable team of experienced people that will be available for the 3 to 4 year task? (The questioner asks NRC to consider the need to seek innovative authorities to provide incentives to attract, retain and provide rewards for a dedicated license review team, including consulting with private sector compensation specialists, to address the "graying"of the NRC workforce, and avoid the typical civil service approach to put the most senior people in the top positions, and after a few years at the pinnacle, they retire.
A: The NRC and its contractor, the Center for Nuclear Waste Regulatory Analyses (Center) will be fully prepared to conduct an objective and effective review of a possible license application for a geologic repository by the U.S. Department of Energy. As part of the ongoing budget and planning process, NRC and the Center are working to ensure that both organizations have the appropriate skill mix and resources necessary to conduct the license application review and prepare a "Safety Evaluation Report". These resource requirements are reflected in the FY 2005 budget currently under preparation, and will also be reflected in the FY 2006 budget. In addition, the project staff across all disciplines at both NRC and the Center are undergoing training to prepare them for this important licensing activity. Recruitment of new staff with appropriate experience and skills are pursued as part of NRC’s routine recruitment efforts. Finally, a new group designated as "High-Level Waste Business and Program Integrator" was created in the Office of Nuclear Materials Safety and Safeguards or NMSS, a Director for this group was appointed in March 2003, and staffing to support the Director is in progress. The Integrator will ensure the High-Level Waste "business" needs are met.

Q: When will the Licensing Support Network (LSN) regulatory guide revision be issued?
A: Currently, it is expected that Revision 1 of Regulatory Guide 3.69 pertaining to Topical Guidelines for the LSN will be issued this summer.

Q: What do you see as the NRC’s on-site representatives’ role during this pre-application period?
A: The on-site representatives (ORs), located in Summerlin, Nevada, will continue their current role during this pre-licensing period. The ORs maintain information associated with the potential high-level radioactive waste geologic repository at Yucca Mountain. Their primary responsibilities include: (a) acting as the NRC's point of contact for prompt information exchange and consultation; and (b) identifying preliminary concerns with potential licensing issues and relating that information to the NRC management and staff. It is noted that the ORs currently do not possess direct safety oversight responsibilities, however they interact with the Department of Energy (DOE) on a daily basis. The NRC ORs' interactions with the DOE are conducted in accordance with a public agreement. This agreement helps the NRC prepare to independently review and judge the soundness of a potential license application. Bimonthly reports are published on the activities of the NRC on-site representatives. These reports are available at this site from January 2000 to present. A poster and information sheet explaining the responsibilities and duties of the on-site representatives are available on the following web sites:
http://www.nrc.gov/who-we-are/locations/hlw-office.html
http://www.nrc.gov/waste/hlw-disposal/public-involvement/on-site-rep.html

Q: Are there any plans to increase the number of NRC on-site representatives, during the license application preparatory phase?
A: The decision whether or not and, if so, when to increase the number of on-site representatives prior to receipt of the license application is under evaluation, as part of the FY 2005 budget currently under preparation. Final decisions will depend, in part, on NRC’s appropriated High-Level Waste budget.

Q: Who at NRC is responsible for GTCC waste planning and disposal? Who does NRC believe is their counterpart at DOE?
A: Responsibility for GTCC planning at NRC:

Environmental and Low-Level Waste Section
Environmental and Performance Assessment Branch
Division of Waste Management
Office of Nuclear Material Safety and Safeguards
U.S. NRC
Contact: Scott C. Flanders, Chief
email: scf@nrc.gov
phone: 301-415-6638

Responsibility for GTCC low-level waste currently managed by DOE:
Office of Environmental Management
U.S. Department of Energy
Contact: Robert Campbell, Program Manager
email: robert.campbell@em.doe.gov
phone: 678-567-0336

RIC 03 Panel Summary Form	Questions/Take-aways from T4 / "Spent Fuel Storage, Transportation, and Disposal"

Panel Chairman W. Richard Borchardt, U.S. NRC:

Q: There is a need for more focus on community impact and the means of translating the technical issues to address present and future community concerns with regard to health/safety/security of an independent spent fuel storage installation (ISFSI). Discussion should address local ISFSI communities with regard to matters of local, regional and national concern. A particular concern is how this would occur when the ISFSI is the only operating facility on a given site.
A: This question has been tabled as a suggestion for consideration in future RIC meetings by the RIC organizing committee from the NRC.

RIC 03 Panel Summary Form	Questions/Take-aways from T5 / Commissioner Dicus

Q: Why does it take so long to address a petition for Rulemaking? In understand that there are some real old ones.
A: The NRC is currently assessing 26 petitions for rulemaking, six of which are greater than five years old. In general, petitions that are denied by the Commission are closed within two to three years. Petitions for rulemaking that are submitted with a strong technical basis to support the requested regulatory action, and are granted in whole or in part by the Commission, are closed with the issuance of a final rule within two to three years. Petitions that are granted in whole or in part, but are not submitted with an adequate technical basis can take several months or even years to complete because the staff must develop the technical basis to support the rulemaking. This technical basis development may require that the Office of Nuclear Regulatory Research (RES), or one of its contractors, perform studies or confirmatory tests. This work by RES or the DOE labs must be prioritized along with other work items within these organizations and may sometimes be deferred for several months or a few years. Once the technical basis is completed, the actual rulemaking may be deferred for several months or years because the rulemaking effort must also be prioritized along with other rulemaking efforts within the agency. Thus, a few petitions for rulemaking that are recognized to have some technical merit but with only a marginal safety benefit may not be closed in a timely manner under the current process.

The interoffice NRC Rulemaking Coordinating Committee has directed that a working group be formed to develop recommendations to improve the NRC's process for addressing petitions for rulemaking. The working group plans to provide a recommendation by the end of FY04.

RIC 03 Panel Summary Form	Questions/Take-aways from T6 / NEI Luncheon

None.

RIC 03 Panel Summary Form	Questions/Take-aways from T7 / Commissioner McGaffigan

None.

RIC 03 Panel Summary Form	Questions/Take-aways from T8 / New Reactors/ Advanced Reactors

R. W. Borchardt

Q: The USNRC has debated elevating core damage frequency, large early release frequency to safety goals; will this be adopted over the next 10-20 years? If not - why is this?
A: In the Staff Requirements Memorandum dated April 16, 2001 for SECY-01-0009, Modified Safety Goal Policy Statement, the Commission disapproved issuance of a revised Reactor Safety Goal Policy Statement until such time as further progress has been made on the agency's various risk-informed regulatory initiatives.

Since the decision in 2001, the staff has continued to make progress on the agency's various risk-informed regulatory initiatives. At this time there is no expectation that the subsidiary objectives (i.e., CDF and LERF) will be elevated as NRC safety goals in the next 10 to 20 years.

Neill Howey

Q: How can IDNS say the "Regulatory Infrastructure" is incomplete for ESP? 10 CFR Part 52 has been in existence for some time. It specifies governing regulations, along with Part 50. Regulatory guidance is what the name says – guidance. The governing regulatory requirements are set for in NRC regulations.
A: This is a good question. Keep in mind that this response is in a time-line that will have at least two of the three ESP applicants submitting their applications for staff review in one month's time. Additionally, once issued, an ESP is an independent licensing action, the NRC can not change. IDNS was asked to bring a state perspective to the RIC panel. If one of the pilot applicants wasn't from Illinois, IDNS would have let the process play out and not gotten too involved. But since one of the applicants is from Illinois, one logically might ask what criteria the applicant used to fill out their application, and what form will it take?

The questioner accurately noted that part 52 has been in existence for over a decade. Plus, it has been revised along the way. It is in fact under major revision now. NRC staff just recently told the industry that there are some policy questions they need to send to the commissioners in a SECY for resolution. There are twenty-two generic issues the industry identified about the Part 52 process; only a handful have reached final agreement. So there are some non-trivial issues concerning the existing process. For a regulation that has been in existence for a long time, it is apparent that many issues were not well thought out. So it caused IDNS to question which Part 52 the applicants used; the out-dated one on the books now, or the draft revised one the commissioners have yet to approve [*NRC Clarification: It is anticipated that the Part 52 regulations now in effect will be applicable to the pilot ESP applicants]* .

In addition, the questioner correctly noted that Part 52 relies heavily on Part 50. We might also add Part 100 for accident dose considerations. However, the industry proposes to circumvent or re-interpret major parts of Part 50 that ask for or require design specific information. The whole idea of a Plant Performance Envelope replacing specific design criteria constitutes a major departure from Part 50 as it exists now. Although not forbidding it, neither the draft Part 50, the existing Part 50 [*NRC Clarification: Part 52 not Part 50 is the applicable regulations for ESP applications*], or the proposed draft SRP mention a PPE type option specifically [*NRC Clarification: The ESP Review Standard RS-002, Processing Applications for Early Site Permits: Draft for Interim Use and Public Comment, issued 12/23/02 does address on page 10 the Plant Parameter Envelope (PPE) approach. Further, NRC Letter from J. Lyons to R. Simard, NEI dated February 5, 2003, explicitly states that ESP applicants may use the PPE approach as a surrogate for facility information to support the required safety and environmental reviews (ADAMS Accession No. ML030230071).*] Hence, our comments that we don't think this is a very firm regulatory structure.

In terms of guidance, I would offer that regulations are regulations, whereas guidance offers acceptable ways to comply with the regulations. There are no NRC endorsed guidance documents, draft or otherwise, to guide an applicant through the application process. The applicants are apparently using an NRC draft revision to a SRP chapter (RS-002) that won't be completed until at least December, and an NEI document that is also under revision, in trying to anticipate what the NRC will use to review the pilot applications. We don't think this is a very firm guidance structure either. We doubt if the NRC would allow a similar approach to the application review for Yucca Mountain, or other major licensing initiatives, under similar circumstances.

In Commissioner Dicus' speech, she said that one of the changes the NRC made over her seven years of service was a departure from the "bring me a rock" regulatory mentality. Tony Pietrangelo of NEI spoke of regulatory coherence as being logically ordered or integrated, united in principles, and systematically connected or consistent. We do not think either applies in the ESP pilot program case.

A couple of things brings IDNS some amount of comfort. First, it is anticipated that the review process will take thirty months or more [*NRC Clarification: Staff schedule for ESP application completion is 21 months for staff review followed by 12 months for the mandatory hearing and Commission decision*]. This probably will result in many RAI's (which the industry normally loathes). Second, the draft revised Part 52 (if issued as proposed) causes the holder of an ESP to revise the emergency planning parts of their ESP at the COL stage when a specific design is chosen. Accident analysis, emergency planning, and emergency response are IDNS functions, and are what got us interested in the ESP process to begin with. These are the reasons, from a state's perspective, why we don't think the "regulatory infrastructure" under the Part 52 process is very firm.

RIC 03 Panel Summary Form	Questions/Take-aways from T9 / Operating Experience Assessments

Q: "How is PRA or other probabilistic analysis used in your effort to evaluate operating experience? How would you characterize the staff's capability and their analytical tools to apply these tools in evaluation of operating experience?
A: All nuclear power plant operating experience events are screened to determine their safety significance and generic applicability. If an event is amenable to being analyzed from a probabilistic perspective, the staff conducts such analysis using an in-house PRA software as well as plant-specific PRA model. There are trained staff members in the operating experience group as well as in other groups within the Office of Nuclear Reactor Regulation, Office of Research, and the Regional offices who are able to perform these probabilistic analyses. The NRC suite of software (Systems Analysis Programs for Hands-on Integrated Reliability Evaluations, aka SAPHIRE) and plant-specific PRA models (Standardized Plant Analysis Risk, aka SPAR models) have been developed and updated for staff's use in conducting PRAs as well as probabilistic events analyses for plant initiating events and plant condition assessments.

Q: "What are the phases of the O.E. Task Force?"
A: There are three phases defined in the Action Plan which addresses the recommendations of the Davis-Besse Lessons Learned Task Force (LLTF). The Operating Experience Task Force effort covers both the Objective Phase (Part I) and most of the Assessment Phase (Part II) of the Action Plan. These are:

1) In the Objective Phase, the Task Force will identify desirable agency operating experience program objectives and attributes in coordination with various internal user and support organizations. The Task Force will also solicit input from the external stakeholders as necessary.

2) In the Assessment Phase, the Task Force will define functional needs to meet the program objectives and attributes, perform gap and overlap analysis, and develop draft and final reports to the Steering Committee recommending specific program improvements and their bases.

The Task Force work is a prerequisite to the staff proceeding with the Implementation Phase (Part III) of the Action Plan, which contains the remainder of the High-Priority LLTF recommendations associated with operating experience. The LLTF recommendations within the Action Plan that are not focused on operating experience are outside the scope of this Task Force effort.

Q: A couple of years ago, hydrogen explosion occurred at the BUR in Japan and in German. There is few activities regarding this event in US, although, it seems safety significant. Why?
A: There are several ongoing activities by both the industry and the NRC regarding this event.

For example, the NRC issued IN 2002-15, "Hydrogen Combustion Events in Foreign BWR Piping," on April 12, 2002, to inform Licensees about the hydrogen combustion events at the BWRs in Germany and Japan. The NRC requested that recipients review the information and consider actions, as appropriate, to avoid similar problems in U.S. plants.

In addition, at a Boiling Water Reactor Owner's Group (BWROG)/NRC senior management meeting on July 25, 2002, the NRC staff asked the BWROG to submit a written report about the group's followup activities in response to the foreign events. The BWROG submitted the information to the staff in a letter dated December 20, 2002 (ADAMS Accession No. ML023610269).

At the February 2003 BWROG/NRC senior management meeting, the BWROG said it will provide a final summary report to the NRC later in 2003 to document the results of the implementation of the BWROG guidance on this issue.

An Information Notice Supplement is expected to be published within the next month which will provide a summary of the actions taken by the BWROG and NRC in response to the hydrogen combustion events in foreign BWRs initially described in IN 2002-15. The NRC staff is continuing to actively monitor the activities relating to hydrogen accumulation.

Q: "The INPO "event database" contains a useful search engine that allows sorting of data via multiple criteria. Would NRC consider adopting such engine to search its data?"
A: The NRC is currently in the initial stage of developing several integrated data systems to evaluate various groups of data associated with operating experience. The NRC plans to make the multiple components of the integrated data system readily retrievable through robust search capabilities.

RIC 03 Panel Summary Form	Questions/Take-aways from T9 / Operating Experience Assessments

Q: "To some extent, every "new" O.E., even minor ones, could be seen as a potential problem with O.E. analysis and communication. Is there any sort of evaluation done to probe this potential on a routine basis?"
A: Each operating experience is routinely evaluated for its safety significance, generic implications, potential for risk impact, and the need to communicate the experience to internal and external stakeholders. A significant level of judgement is applied in this evaluation. An integral part of the current task force effort is develop performance measures to evaluate the effectiveness of the operating experience program. It is currently not done on a routine basis.

RIC 03 Panel Summary Form	Questions/Take-aways from T10 / Fire Protection

Q: For a dual-unit site, current regulatory guidance explicitly states that coincident accidents on both units does not have to be assumed. Guidance also states that coincident fires on both units do not have to be assumed. Does a coincident fire on one unit with an accident on the other unit have to be assumed?
A: Coincident fires on one unit with an accident on the other unit are not required to be assumed by regulation. Typically worst case fires are considered for evaluation. Although coincident fires are not assumed, single fires that may affect multiple units are considered. Also, coincident accidents are not considered, yet initiating events that may affect multiple units are considered, e.g., seismic events.

Q: In enforcing GDC-3 [10 Code of Federal Regulations (CFR), Part 50, Appendix A, General Design Criteria (GDC) 3 - Fire Protection], how does NRC distinguish between fire and explosive hazards?
A: 10CFR Part 50.48 is the Fire Protection Rule. It is 10CFR Part 50.48 that requires each operating nuclear power plant to have a fire protection plan that satisfies GDC-3 (Section (a)(1)). Compliance with 10CFR Part 50.48 is met through the use of a fire hazards analysis of plant fire areas (Sections (a)(1)(i)-(a)(1)(iv) and (a)(2)(iii)).

The fire protection and explosion hazard guidance depends on the date of the license for each plant. If the NRC issued a construction license prior to July 1, 1976, Appendix A to Branch Technical Position (BTP) APCSB 9.5-1 (ADAMS Legacy Library Accession No. 8712210199) applies. For later plants the NRC provided guidance in Section 9.5.1 of NUREG-0800, Standard Review Plan.

Due to the nature of the combustibles in the plant the Fire Protection Rule assumes that fire is the primary hazard. The NRC included specific guidance for explosive hazards where applicable in nuclear power plants. The following are examples of the treatment of explosive hazards (Appendix A to the BTP APCSB 9.5-1):

D.1(h) Separation of buildings containing safety related (SR) systems from oil filled transformers.
D.2(b) Restrictions regarding bulk gas storage.
F.7 Protection of station battery rooms against fire explosions.
G.1 Storage of welding and cutting gas systems.

RIC 03 Panel Summary Form	Questions/Take-aways from T10 / Fire Protection

Q: Hydrogen was first identified as an explosive hazard at Main Yankee in 1978. Provide a history of how NRC has resolved hydrogen as an explosive hazard in turbine-generators, considering the large aircraft crash terrorist act. Consider contents of hydrogen in turbine generators and in nearby storage tanks and batteries."
A: The NRC has considered hydrogen as a hazard at nuclear power plants for many years. Beginning in 1977 the Standard Review Plan (SRP) 0800, Section 9.5.1, Fire Protection, provided guidance for hydrogen storage. Also, Appendix A to Branch Technical Position 9.5-1, contained guidance for hydrogen storage. Generic Letter 93-006, "Research Results on Generic Safety Issue 106, 'Piping and the Use of Highly Combustible Gases in Vital Areas'," addressed the piping and use of highly combustible gases in vital areas. Also, Information Notice 01-012, "Hydrogen Fire at Nuclear Power Station," provided information regarding a specific fire at one nuclear power plant.

Turbine generators are typically located remotely from safety related equipment. NUREG-0800, Standard Review Plan, Section 9.5.1, from July 1981, included guidance that turbine buildings should be separated from adjacent structures containing safety related equipment by a three hour rated fire barrier. A hydrocarbon fire has generally been considered the greater hazard in turbine generators rather than a hydrogen explosion, due to the large quantities of lubricating oil. The fact that turbine generator hydrogen is generally separated from safety related equipment is the fundamental mitigating strategy with respect to turbine generator fire or explosion.

With regard to resolution of hydrogen as an explosive hazard in turbine generators, considering a large aircraft crash, licensees have implemented certain actions as a result of NRC advisories and Orders to mitigate the effects of a September 11-type aircraft attack. In January 2002, the NRC completed an initial assessment of the scope of nuclear power plant vulnerabilities to aircraft attack. More detailed analyses are continuing this year and a final report is expected in December 2003. The agency will not comment on the conclusions of any vulnerability assessment. The NRC will not make an unclassified version of it's assessment available. [From Questions and Answers Related to Nuclear Power Plant Vulnerability to Aircraft Attack].

Hydrogen storage tanks are not typically located near turbine generators. Hydrogen storage tanks are generally located outdoors away from the plant and other safety related tanks and equipment. The staff recently completed a survey and a Temporary Instruction (TI) 2515/146, "Hydrogen Storage Locations," that verified that reactor sites were in compliance with their commitments for hydrogen storage locations.

Although station batteries do not contain hydrogen, they do have the capacity to generate hydrogen. Battery rooms are equipped with ventilation systems and hydrogen concentration sensors to ensure that explosive environments are not created.

Q: Suggestion - If web is too vulnerable for plant nuclear info, why not place info into PDR & control access? Refer website viewers to PDR.
A: In addition to initial concerns about the possible sensitivity of plant-specific information available on the NRC web site following September 11, 2001, the decision to remove some nuclear plant information from the site resulted from the re-design of the web site and the recognition that much of the information had not been maintained. The NRC recently restored pages providing information on each operating nuclear power plant. In addition, our web site directs users to our electronic recordkeeping system, ADAMS, and the PDR for information not directly available from the web site. Some documents, such as plant final safety analysis reports (FSARs), require review for possible sensitive information before they will be released by the PDR staff. We are continuing to review documents to identify possible sensitive information and have generally withheld little information that had been previously available to the public. The NRC staff is preparing guidance for internal use and for use by licensees in an effort to remove sensitive information from documents such as the FSARs so that they may be made freely available to the public. If you have any additional questions or suggestions regarding how we might provide information via the NRC web site, ADAMS, and PDR, please feel free to use the "contact us" feature on our web site.

RIC 03 Panel Summary Form	Questions/Take-aways from T11 / International Issues & Perspectives

None.

RIC 03 Panel Summary Form	Questions/Take-aways from T12 / Commissioner Merrifield
None.	

RIC 03 Panel Summary Form	Questions/Take-aways from T13 / International Experience

Q: How safe are the VVER's? How does their safe operation compare with other country's PWR's?
A: There are no Soviet designed VVER reactors operating in the U.S. Most of the world's VVER reactors were built and now operate in Russia, Ukraine and what was formerly the Soviet bloc. For insight on the safety of VVERs and how their safety operation compares with PWRs of other countries, you may wish to contact:

Nuclear & Radiation Safety Department, Gosatomnadzor of Russia
34, Taganskaya Street
Moscow, 109147
Russia

Q: Why did Dominion withdraw from MOX fuel program? Is this withdrawal permanent?
A: The NRC recognizes the sovereign decision-making authority of individual U.S. utilities and the sensitivity of its business decisions. For this reason, you may wish to contact:

Director, Nuclear Safety
North Anna Power Station
Dominion Generation
Post Office Box 412
Mineral, VA 23117

Q: What has been France's "accident" rate?
A: For information on France's "accident" rate, you may wish to contact:

Nuclear Installations Safety Directorate (DGSNR)
99 rue de Grenelle
75353 Paris 07 SP
Ministry for Economy, Finance and Industry
Secretariat of State for Industry
Ministry for Territory Development and the Environment

RIC 03 Panel Summary Form	Questions/Take-aways from F1 / Region II Breakout
None.	

RIC 03 Panel Summary Form	Questions/Take-aways from F2 / Region III Breakout

Q: Can the alternate dispute resolution (ADR) process be utilized for SDP issues to save NRC and licensee resources?
A: The agency is about to seek Commission approval for a pilot of ADR in enforcement. Based on the results of that pilot, the NRC may expand the use of ADR in enforcement and could look at its use in other areas such as the SDP.

Q: Review (of) licensee data (reveals) that R3 issues more 10CFR 50.9 violations than other regions.
A: Region III Enforcement and Investigation Coordination Staff collected 10 CFR 50.9 data from 1999 through March 2003 for all the regions. The following results include escalated and non-escalated cases: Region I - 13; Region II - 10; Region III - 10; and Region IV - 3.

Q: The Regulatory Impact Feedback forms/meetings may not be the most effective tool in eliciting feedback on inspector or regional implementation of ROP.
A: These forms are intended to be used by regional management as an outreach tool to proactively seek and capture constructive feedback from our licensees. Each manager that visits a site should seek this feedback as prescribed in our Inspection Manual. The tool comes with the clear expectation that NRC management will accept such feedback in the spirit in which it is offered and seek to improve our program because of it. A professional relationship that fosters clear communication must be the objective of both parties, and in fact, is the expectation of all stakeholders. In discussion with the NRR Program Office, the region can not think of any substitute tool that replaces effective communication between NRC and licensee management.

RIC 03 Panel Summary Form	Questions/Take-aways from F3 / Region IV Breakout

Q: FOCI Point of Contact in NSIR
A: INFOSEC/NSIR (Keith Everly) is responsible for granting the Facility Security Clearance of which the FOCI determination is a component. The other component is completion and certification of a Standard Practice Procedures Plan. INFOSEC provides the status of the FOCI reviews, the ADM/Security Branch (Michael W. Bodin) provides the status of the individual security clearances.

Q: Where do we stand industry-wide in getting clearances approved for designated licensee employees & who can tell us our individual status?
A: Information regarding the status of clearances should be requested from the Division of Facilities and Security/Security Branch. The NRC has issued 6 clearances as of April 25, 2003. Clearances cannot be issued until the FOCI has been approved. INFOSEC/NSIR (Keith Everly) is responsible for the FOCI requests. The Division of Facilities and Security/Security Branch (Michael W. Bodin) can be contacted directly by the licensee for status updates or an explanation of the process for clearing individuals.

RIC 03Panel Summary Form	Questions/Take-aways from F4 / Region I Breakout

None.