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July 26, 1999

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Ralph E. Beedle

December 29, 1999

Mr. David L. Meyer
Chief, Rules and Directives Branch
Division of Administrative Services
Office of Administration
U.S. Nuclear Regulatory Commission
Mail Stop T-6 D59
Washington, DC 20555-0001

SUBJECT: Public Comments on the Pilot Program for the New Regulatory Oversight Program

Dear Mr. Meyer:

On behalf of the nuclear energy industry, the Nuclear Energy Institute is submitting the enclosed comments on the Pilot Program for the New Regulatory Oversight Program, published in the *Federal Register* on July 26, 1999 (64 Fed. Reg. 40394).[1]

NEI appreciates NRC's continuing efforts in developing the new Regulatory Oversight Process. The progress made during the pilot program would have been impossible without the degree of public interaction and cooperation exhibited by all stakeholders. Without forsaking its responsibility to make final decisions, NRC was willing to openly share its ideas and allow public comment on a real-time basis. The result is a far better product than could have been achieved in the past. This new paradigm of communication and understanding between the regulator, licensees and other stakeholders is to be commended. It should also be emulated for future regulatory improvement initiatives.

The industry comments are arranged in five enclosures. Enclosure 1 provides specific comments on the information requested in the request for public comments. Enclosure 2 addresses Performance Indicators. Enclosure 3 addresses Inspections. Enclosure 4 addresses the Significance Determination Process. Enclosure 5 addresses the NRC public website.

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[1] The comment period was extended to December 31, 1999 in the *Federal Register* on November 4, 1999 (64 Fed. Reg. 60244).

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PDR-RegP NEI

The enclosures provide detailed comments on many issues. However, there are several issues that must be addressed prior to full industrywide implementation of the program:

1. The issue of how 10 CFR 50.9 is to be applied for inadvertent errors in the submittal of performance indicator data must be resolved prior to full implementation. It is recommended that there be a period during which data errors and technical questions caused by differing plant configurations and processes be resolved without any enforcement action. This may take about a year for all the issues to emerge and be resolved. After a sufficient learning period, the industry position is that minor errors that do not cause the indicator to cross a threshold are not material because they do not affect NRC actions. Therefore, these errors would not be violations. Errors that do cause the Green-White threshold to be crossed result in a minor increase in risk and a minor increase in NRC action. In this case, a violation has occurred, but could be treated as a minor violation or a non-cited violation, unless the criterion for issuing a Level IV violation as described in the interim enforcement policy applies. Crossing two thresholds, on the other hand, is a major error, and may warrant a Level III citation if NRC was not aware of the condition until the error was corrected.

   A related issue is the time interval for reporting performance indicators to the NRC. While the pilot plants were able to submit data within 14 days of the end of each reporting period, most of the pilot plants found it necessary to devote additional resources to assure the quality of the data. The time interval for reporting data should be extended in light of the emphasis being placed on accuracy. The industry recommends that the interval be extended to match the time the staff estimates it can publish inspection reports following completion of an inspection, thereby ensuring that both PI and inspection results reflect the same time period on the public website.

2. The Security Equipment Performance Index performance indicator is deficient and must be corrected. The Green-White threshold for this indicator is overly conservative and is not based on historical performance. The threshold for this indicator, as calculated, would require an availability for individual security equipment which exceeds that required for the emergency diesel generators and other safety systems, even though the unavailability is fully compensated by the guard force. The White-Yellow threshold for this indicator is inappropriate. The Yellow band for the performance indicators is meant to represent a significant reduction in safety margin. Since unavailable security equipment is fully compensated by the guard force, a significant reduction in margin cannot occur. Note that several other PIs do not have Yellow or Red bands because the safety significance of

the indicators cannot be established consistent with other indicators in the program (e.g., unplanned power changes and safety system functional failures). This indicator requires additional analysis and correction prior to full implementation of the program.

3. The Security Significance Determination Process is deficient and must be corrected. The security SDP lacks sufficient guidance to generate repeatable results, and overemphasizes situations in which there is no significant increase in the likelihood of damage to the reactor core, making it inconsistent with the other PI thresholds and SDP findings. The guidance for the security SDP should be improved for full implementation. The industry has proposed a revision to the Security SDP that provides a method for generating consistent outcomes and appropriately determining the safety significance consistent with the other cornerstones.

4. The thresholds for performance indicators and SDP results need to be reviewed and made consistent across the cornerstones. For the Action Matrix to work as envisioned, a White, Yellow or Red input needs to have the same meaning in terms of safety significance for all cornerstones. Currently, some of the possible outcomes in the Emergency Preparedness and Security cornerstones are not consistent with the outcomes in the reactor safety and Radiation Protection cornerstones.

With the resolution of the above issues, NEI believes that the new oversight process can be successful in achieving its goals to:

- ensure that nuclear power plants continue to operate safely;
- improve NRC efficiency by focusing resources;
- reduce unnecessary regulatory burden on licensees; and
- enhance public confidence in the safe operation of nuclear power plants.

The industry notes that some NRC staff is concerned over the treatment of cross-cutting issues in the new process. A fundamental tenet of the revised reactor oversight process is that cross-cutting issues will manifest themselves in departures from expected norms of performance, thereby causing the established thresholds for PIs and inspection findings to be exceeded. While this tenet should be validated during industry-wide implementation, it is premature and inappropriate to incorporate subjective judgments into the process absent any performance issues. If the validation concludes that cross-cutting issues need to be a more structured part of the overall process to meet process objectives, formal changes to the process should be pursued to incorporate cross-cutting issues as an integral part of the program. It is important to note that both industry and NRC have recognized the importance of the cross-cutting issues and have taken steps to address them. For example, the NRC assigns approximately 15 percent of its baseline inspection hours focusing on the cross-cutting issues. The industry, through the Institute of Nuclear Power Operations (INPO), has recently completed a principles document for effective self-assessment and corrective action programs that captures best practices from within the industry. We believe these measures will provide adequate attention to cross-cutting issues.

We recognize that further refinements to the process will be necessary as industrywide participation identifies additional issues. However, the industry believes that sufficient progress has been made to support industrywide implementation beginning in April 2000.

Sincerely,

Ralph E. Beedle

Enclosures

ENCLOSURE 1

NEI Comments On NRC Federal Register Notice Requesting Public Comment On The Pilot Program For The New Regulatory Oversight Program

1. **Does the new oversight process provide adequate assurance that plants are being operated safely?**

   NEI believes that the combination of performance indicators (PIs) and safety focused inspection activities address the seven safety cornerstones, and provide a solid basis on which to provide adequate assurance that nuclear plants are being operated in a safe manner. The combination of performance indicator thresholds and Significance Determination Process (SDP) evaluation thresholds provide appropriate triggers for increased regulatory involvement, through supplemental inspections, well ahead of risk-significant performance degradation. If utility performance continues to decline below expected norms, the Action Matrix will provide NRC with an effective tool to focus increasingly detailed supplemental inspection to ensure utility management is addressing the underlying causes and achieving effective corrective action.

   The performance indicators are relevant and appropriate to use in safety assessment based on industry and NRC analysis. The PIs are considered to be indicators, not measures of performance. As such, they provide an indication to licensees and the NRC about performance. If the indicator shows declining performance, more detailed analysis will be performed to determine the cause and implement corrective action. The indicators were chosen for the most part on the basis of readily available data which was already being collected. They do not cover all areas of safety (that is the purpose of the new baseline inspection program) and they can be improved over time. However, with baseline inspection, they do provide a robust picture of nuclear plant safety.

   In most cases historical data was available to assess the indicators to ensure that the data could be collected and applied in assessing performance. The performance indicator thresholds are appropriately based on historical data and the pilot plant program, with one exception, the Security Equipment Performance Index. The Green-White threshold for this indicator is overly conservative and is not based on historical performance. The threshold for this indicator, as calculated, would require an availability for individual security equipment which exceeds that required for the emergency diesel generators and other safety systems, even though the unavailability is fully compensated by the guard force. The White-Yellow threshold for this indicator also requires rethinking. The Yellow band for the performance indicators is meant to represent a significant reduction in safety margin. Since unavailable security equipment is fully compensated by the guard force, it is not possible to have a significant decrease in margin. Note that several other PIs do not have Yellow or Red bands because the safety significance of the indicators cannot be established consistent with other indicators in the program (for example, unplanned power changes, and safety system functional failures). This indicator requires additional analysis and correction prior to full implementation of the program.

In general, the industry has found that the scope and frequencies of the baseline inspection procedures are adequate to address their intended cornerstones. A possible exception is the Fire Protection inspection module and its SDP. At this time, two pilot plant fire protection inspections have been conducted with one being completed (Salem). The fire protection SDP is very subjective in evaluating the degradation of fire protection defense in depth features. This requires a common understanding between the NRC and the licensee on the overall impact that these defense in depth features have on increasing the risk significance of the inspection finding. The fire protection SDP should be further reviewed for effectiveness.

The pilot plants' experience with the SDP indicates that it provides conservative results. However, the security SDP lacks sufficient guidance to generate repeatable results, and overemphasizes situations in which there is no significant increase in the likelihood of damage to the reactor core, making it inconsistent with the other PI thresholds and SDP findings. The guidance for the security SDP should be improved for full implementation. The industry has proposed a revision to the Security SDP that provides a method for generating consistent outcomes and appropriately determining the safety significance consistent with the other cornerstones.

NEI understands that NRC is in the process of developing additional performance indicators and several additional SDPs (for example, shutdown risk and containment). The process used to develop the current assessment tools should be applied for any additional ones. That process involved significant effort and analysis. For the PIs, the key steps included: identifying a potential indicator (including determining whether the indicator would provide significant additional risk insights beyond the current indicators or could be used to substitute for current inspection), developing a definition, collecting historical data, establishing thresholds, pilot testing the indicator, and training industry and NRC on the indicator. In the SDP area, similar analysis and care is necessary to ensure a worthwhile assessment tool, including table-top exercises and a pilot program prior to implementation. The containment SDP should be based on the risk insights contained in NUREG-1465, New Source Terms. In that NUREG, NRC identified that potential radiation releases have time dependencies and chemical forms different from those currently assumed in plant designs. The NUREG insights are better suited to define the safety significance of any potential containment finding.

2. **Does the new oversight process enhance public confidence by increasing the predictability, consistency, clarity and objectivity of the NRC's oversight process?**

NEI believes that the new oversight process results in NRC assessments of licensee performance and NRC actions that are more understandable, predictable, consistent, and objective. The process is clearly described in NRC management directives and procedures and is based on objective data and results, not subjective opinions. The safety results are made available on the NRC website. The interested public can review this data and the NRC's activities. The result should be an increase in public confidence in the NRC's ability to regulate the nuclear utility industry, and that the plants are being operated in a safe manner.

NEI and the pilot plants have found that the new inspection reports follow the new process guidelines, which have a more understandable format. However, NEI noted some inconsistencies in the reports with respect to use of terminology, identification and discussion of findings, and level of detail. . There has been confusion on the use of the words findings, minor violations, etc. NEI believes that the additional guidance provided to the Regions regarding treatment of minor violations, minimum thresholds for screening potential findings, report content and the recent enforcement guidance memorandum (EGM 99-006) will help address the inconsistencies noted during the pilot program. It is essential to the program that consistent methodologies and terminology be used nationwide. NEI recommends that NRC continue its national overview of finding determinations, performance indicator questions, and inspection reports to ensure consistency across the four NRC Regions and consistent, fair and equal treatment for licensees.

The NRC web page is generally very usable. The NRC reactor oversight web page accurately reflects performance indicator data. However, NRC needs to clarify whether a finding or the violation(s) associated with a finding are to be presented on the screen. This difference is important because findings, and the number of findings (white, yellow or red) are used in the action matrix to determine NRC supplemental action. The presentation of the associated violations on the web page rather than just the finding can lead to confusion with application of the action matrix. NEI would suggest that NRC establish that just the single finding be presented on the web page. A description of the underlying violation(s) can be found in the body to the associated inspection report. Similarly, NRC needs to clarify how findings should be reported when the hardware problem existed on only one unit, but programmatically it might have affected other units. NEI would suggest that NRC assign the finding to only the unit(s) that experienced the hardware problem.

While the current explanation of performance indicators is excellent, NRC should continue to improve the material as necessary to make it clear to the public. For example, some initiating event and mitigating system PIs which are based on a rate can cross thresholds even when there are no changes in measured data in the current quarter. The reason is that the indicator is a rate (for example, number per 7000 critical hours) and if the current quarter had relatively few critical hours compared to the quarter it replaced in the rolling count, the PI will necessarily increase and could cross a threshold, even without an increase in the measured parameter. NRC should develop a standard reporting note that could be used to identify when such things happen to avoid over reaction by NRC or misunderstandings by stakeholders.

3. **Does the new oversight process improve the efficiency and effectiveness of the regulatory process focusing agency resources on those issues with the most safety significance?**

NEI and the pilot plants believe the answer to this question is yes. The NRC has attempted to answer the question during the pilot program by trying to determine whether the new process uses less inspection hours. The statistics will be determined from the pilot process; however, this metric is probably not appropriate to answer the question of efficiency and effectiveness. The amount of time to conduct an inspection will vary depending on what is found during the inspection and the training level and experience of the inspector. The more appropriate questions are whether the procedures focus on safety significant issues and whether they are capable of being consistently applied across regions and within regions. This is a matter of training and communication -- and leadership at the national level -- to ensure that regional or individual inspector peculiarities are managed to ensure that the focus is maintained on safety and risk informed issues. The procedures should be evolutionary, reflecting a learning organization.

The actual amount of time devoted to inspection during the pilot varied considerably from plant to plant and compared to the initial estimates. This is not to be unexpected in a pilot program testing new procedures. In some cases, it appeared that the inspections could have been completed in less time, in other cases more time appeared to be necessary. Based on the pilot plants' experience with the baseline inspection program, NRC management must continue to establish challenging targets for inspection resource requirements. It must also provide sufficient monitoring of inspection activities to ensure that resources are used efficiently.

NEI and the pilot plants have found the new regulatory oversight processes to be more effective. This conclusion is based on the fact that the inspectors now have inspection modules which are more focused on safety issues and equipment, and the assessment process itself encourages inspectors to devote their efforts to potential problems which are more likely to be of safety significance.

By increasing the focus on risk significant attributes, as opposed to procedural compliance (particularly licensee compliance with its own procedures, not NRC regulations) the NRC can reduce its resource load and at the same time provide greater safety assurance. By focusing on results and risk rather than blanket coverage, the NRC can be more resource efficient without decreasing its assurance of safety. Compared to the previous core inspection program, it would seem that the baseline inspection would not necessarily involve fewer inspection hours; however, for a plant with mostly all green findings, there should be less need for region based inspection. Hence the total inspection effort should be less.

The inspection procedures themselves address the cornerstones of the program. They are written in a clear manner that should be capable of being consistently applied across regions and plants. The problem identification and resolution inspection procedure contains procedural detail that seems at odds with a results oriented inspection. Initial inspections in this area appeared too process oriented; the more recent ones seem to have shifted the focus more appropriately to results.

A major improvement in NRC inspection efficiency and effectiveness could be achieved by relying more on licensee self-assessment activities. If the licensee conducts a self-assessment that addresses the same attributes as a NRC inspection module, NRC should be able to reduce its total inspection effort. For example, the NRC could review the scope and results of the licensee self-assessment, and if adequate, need not expend its own resources by redundantly reviewing the same area.

NEI would suggest that NRC consider the following changes to the baseline inspection program to further improve efficiency:

- The frequency of the permanent plant modification module be revised to match the plant refueling outage schedule. It is unlikely that significant modifications will be made to risk-significant systems outside of refueling outages. The current annual frequency is unnecessary.
- The problem identification and resolution inspection can be adjusted to a biennial basis after a satisfactory baseline inspection. Reasonable assurance of the corrective action program integrity will be provided by other inspections that, by design, test on a continuing basis the

effectiveness of the corrective action program. The module should include a provision that the frequency can be readjusted to yearly if a decline in performance is observed.

- Radiation monitors are inspected in the radiological controls modules and the maintenance rule modules. The scope of these two efforts should be coordinated to avoid redundancy. In addition, the safety-risk basis for the inspection efforts in each should be made clear.
- The permanent plant modification module and the 50.59 and SAR change modules could inadvertently review the same design change packages unless these modules are coordinated. NRC should review the module scopes and the planning practices to ensure that there is no redundancy between these two inspection efforts.
- The permanent plant modification module and the safety system design and performance module can also review the same areas. NRC should review the module scopes and the planning practices to ensure that there is no redundancy between these two inspection efforts.

NEI believes that the inspection planning process can be performed in an effective manner to support the assessment cycle. While inspections were rescheduled more frequently than expected due to the nature of the pilot program, it was clear that the workload is manageable. The structured approach outlined in the action matrix will help the planning process by eliminating many of the unnecessary reactive inspections that have been conducted in the past. In addition, for white findings, the action matrix sets an expected time frame for the follow-up inspections (after utility corrective action activities are well underway) that will facilitate better planing and more effective inspection activities.

NRC needs to provide a better definition for what gets treated as a finding and therefore requires a phase II review. Without a clear threshold for findings, the process can become inefficient by requiring evaluations for too many minor issues. For the reactor SDP, NRC has provided additional guidance to the Regions regarding treatment of minor violations and minimum thresholds for screening potential findings. This guidance should be clearly applied to issues in other cornerstones. The Occupational Radiation Safety SDP flowchart has criteria which result in an issue being dropped as a finding. Similar screening thresholds should be considered for the other SDPs. Unlike the Reactor Safety cornerstone where PSA is utilized in Phase III, there is no defined process to involve the License in the significance determination for the Radiation Safety, Emergency Preparedness, and Security cornerstones. For these cornerstones, this needs to be formalized.

Additionally, the terminology of Phase I, II, and III should be consistent across all cornerstones.

4. **Does the new oversight process reduce unnecessary regulatory burden on licensees?**

"Burden" consists of licensee activities necessary to maintain acceptable safety (i.e., necessary burden), licensee activities required by the regulator which do not affect safety but which are imposed on the licensee, fees for NRC activities which are necessary to ensure the licensee maintains acceptable safety, and fees for NRC activities which do not add safety value. The pilot, by and large, has shifted the focus toward a value-added safety focus and away from unnecessary activity. By sharpening NRC's focus on safety outcomes and results and away from insignificant procedural violations, the process is reducing unnecessary burden.

The pilot plants were able to submit PI data by the due date (14 days following the end of the reporting period). However, while the pilot plants were capable of meeting the due date, the question of the industry's susceptibility to 10CFR50.9 sanctions for non-willful errors must be resolved. Another question, which relates to regulatory burden and the value added by a two week submittal time, is when does NRC really need the data in order to plan effectively? The two week period does add a burden, which would lessen if the submittal date were three or four weeks. Note that an extension from two to three weeks would ease the regulatory burden. It would not, however, resolve the question of how errors are treated under 10CFR50.9. The industry recommends that the interval be extended to match the time the staff estimates it can publish inspection reports following completion of an inspection, thereby ensuring that both PI and inspection results reflect the same time period on the public website.

Generally speaking, licensee resources for collecting pre-inspection data and for responding to non-significant issues has been less. There has been an increase during the pilot of resources required for PI collection and general inspection support. NEI and the pilots believe this is due to starting up a new process (for both licensees and inspectors), necessary changes to clarify the PIs, and the compression of the inspection program in order to test a year's worth of inspections during the pilot. Efficiencies in PI collection and in conducting inspections will be gained with experience. Overall, there will be gains in effectiveness and efficiency with an increased focus on safety issues.

NEI identified problems with the process used to evaluate a potential non-green finding (i.e., white or above). The assumptions used in the NRC phase 2 and phase 3 analyses were not documented in a timely manner. The delay hampered the utility's understanding of NRC's perspective on the issue. It also made the interactions with NRC less efficient. NRC should ensure that the assumptions it makes in the preliminary evaluation are properly documented in a timely manner and shared with the utility. NRC needs to formalize and document the process that will be used to issue non-green non-violation findings. Because the process is not documented, the treatment of the non-green non-violation findings during the Pilot Program was inconsistent.

5. **The new oversight process does not currently provide an overall assessment of performance of an individual safety cornerstone other than a determination that the cornerstone objectives have or have not been met. However, it does identify regulatory actions to be taken for degraded performance within the safety cornerstones. Is an overall safety cornerstone assessment warranted or appropriate?**

   No. The assessment of each cornerstone is based on a set of performance indicators and inspection activities which indicate whether all of the cornerstone objectives are being met. These are objective, results-oriented facts which are easily understandable and actionable, based on risk significance. Attempting to somehow combine these different indications into a single overall assessment number or color would create a meaningless abstraction. First of all, it would rely on subjective judgements about the relative importance of different indicators or inspection findings that are measuring different activities or conditions. Second, such an abstraction is not actionable, that is to say, one cannot "fix" a cornerstone. One can fix issues and conditions that are grouped under a cornerstone. NEI believes that the NRC can use the Action Matrix to determine what level of NRC action is appropriate in response to indications of declining utility performance, given the number and extent of conditions outside the normal (Green) band of performance. The NRC staff will need to demonstrate the appropriate discipline to allow completion of the utility response to white findings in order for the oversight process to work as designed.

6. **Licensee findings as well as NRC inspection findings are candidates for being evaluated by the significance determination process. Does this serve to discourage licensees from having an aggressive problem identification process?**

   No, for several important reasons. First of all, the new process is based on safety significance, not legalistic arguments over who found the problem first. It is in the licensee's best interest to aggressively self identify problems and correct them before performance degrades and a threshold is crossed, resulting in increased NRC inspection. This is a strong incentive to have a robust program. Second, licensees are required by the regulations, specifically 10CFR50 Appendix B, Criterion XVI, Corrective Action, to establish measures to assure that conditions adverse to quality are promptly identified and corrected. This requirement does not go away with the new assessment program. Third, the new enforcement policy, by shifting the emphasis away from punitive civil penalties and nuisance violations toward a focus on identifying and resolving safety issues, more than compensates for any so called "discouragement" for licensees to have an aggressive problem identification process. Fourth, it is not a new practice for NRC to review licensee identified deficiencies. In fact, licensee identified problems have often resulted in significant and costly enforcement action, albeit some "credit" was given for self-identification. Fifth, licensees are now operating in a competitive environment, and realize that to remain competitiveness they need aggressive corrective action programs to maintain safety, reliability and profits.

7. **In the new oversight program, positive inspection observations are not included in NRC inspection reports and the plant issues matrix (PIM) due to a lack of criteria and past inconsistencies and subjectivity in identifying such issues. Previous feedback on this issue indicated that the vast majority of commenters believed positive inspection findings should not be factored into the assessment process. Does the available public information associated with the revised reactor oversight process, including the NRC's web page which includes information on performance indicators and inspection findings, provide an appropriately balanced view of licensee performance? If not, should positive inspection findings be captured and incorporated into a process to reach an overall inspection indicator for each cornerstone?**

NEI supports the elimination of all subjective comments, both positive and negative. Inspection results should be based on factual information that characterizes the safety significance of findings consistent with the thresholds established by the program. The thresholds have been set based on expected norms of performance and risk significance. If a licensee is operating in the green band in PIs, and has green inspection findings, i.e., violations of low safety significance, then this in itself is a positive statement. The facts speak for themselves. Subjective statements are inappropriate as they are subject to inconsistency and misinterpretation. It is important, however, for NRC to provide a better definition of what gets treated as a finding. Without a clear threshold for findings, inspectors are free to discuss any issue, even if no violation of regulatory requirements or safety issues are involved. As a result, inconsistencies and subjectivity will be introduced by a discussion of perceived negative issues. NRC should limit the discussion of non-risk significant issues.

NRC may want to consider adding a short summary of the all completed inspections with a web link to convey the scope of the inspection efforts (man-hours and areas inspected). This additional summary information would provide a more balanced view of the inspection for those stakeholders who cannot invest the time in reading the complete inspection reports. It would provide a sense of balance by making clear that NRC did not find any significant issues during its inspection. Finally, NRC might want to make a declarative statement that, in the area under inspection, the plant was being operated safely.

8. **The staff has established several mechanisms such as public meetings held in the vicinity of the plants, this Federal Register Notice, and the NRC's website to solicit public feedback on the Pilot Program. Are there any other appropriate means by which the agency could solicit stakeholder feedback, in a structured and consistent manner, on the Pilot Program?**

NEI believes that NRC should continue its efforts to keep the public informed on its oversight activities and on the continuing safe performance of the nation's nuclear power plants. With the addition of objective performance indicators and inspection findings classified based on risk significance, the NRC has created a mechanism which should enhance the public's perception and acceptance of nuclear power generation. As some of the public outreach effort during the pilot program has shown, NRC needs to continue and perhaps enhance its communication to the public. We believe public meetings to explain the new process should be held at all nuclear plants in the country.

9. **Are there any additional issues that the agency needs to address prior to full implementation of the new oversight process at all sites?**

An important issue that needs to be resolved prior to full implementation of the program is the accuracy of performance indicator data. Reporting accuracy continued to increase during the pilot program, as knowledge of the details of the counting methodology increased, and as questions regarding specific configurations and conditions at individual plants were resolved. By the conclusion of the pilot period, plants were able to report data in an accurate manner such that NRC could take appropriate action in accordance with the Action Matrix.

A system of biweekly meetings between the NRC, pilot plants, NEI and the public were held to discuss questions and obtain clarification of the indicator definitions. In addition, lists of frequently asked questions, which answers were approved by the NRC staff, were posted to the NEI website for the pilot and other plants information and use. However, the pilot FAQ process was not always effective in dispositioning questions in a timely manner in order to support regional inspections. In some cases, interpretations of the performance indicator manual were made by the NRC without dialogue in a public forum with the industry.

The period between January 21 (when the entire industry submits data) and April when the program begins must be used effectively to resolve any other unique questions and clarifications needed to ensure the remainder of the industry will be able to report their indicators in an accurate manner.

An inspection of the performance indicator data at each plant will provide a basis for common understanding and expectations for the plant and its resident inspectors. In discussing "accuracy" it must be kept in mind that there will continue to be some differences of interpretation which need to be resolved on an ongoing basis. These differences in interpretation should be characterized as URIs until an institutionalized process (such as described above) can render a final position. These differences in interpretation should be recognized as such, and not be subject to enforcement action. In addition, minor errors will occasionally occur that will need to be corrected, but as seen during the pilot process, do not negate the ability of the NRC to conduct its oversight function. Therefore, the issue of how to handle errors in data must be resolved before the process can be fully implemented. Resolution must include guidance on enforcement with respect to 10CFR50.9. It is recommended that there be a period during which data errors and technical questions caused by differing plant configurations and processes can be resolved without any enforcement action. This may take about a year for all the issues to emerge and be resolved. After a sufficient learning period, the industry position is that minor errors that do not cause the indicator to cross a threshold are not material because they do not affect NRC actions. Therefore, these errors would not be violations. Errors that do cause the green/white threshold to be crossed result in a minor increase in risk and a minor increase in NRC action. In this case, a violation has occurred, but could be treated as a minor violation or a non-cited violation, unless the criterion for issuing a Level IV violation as described in the interim enforcement policy applies. Crossing two thresholds, on the other hand, is a major error, and may warrant a Level III citation IF NRC was not aware of the condition until the error was corrected.

NEI's experience is that the performance indicator data can be reported accurately such that NRC can take appropriate action; however, a few of the indicator definitions require judgment and interpretation that can lead to errors in reporting. In particular, NEI supports the ongoing effort to establish a standard set of criteria for reporting safety system unavailability between the NRC performance indicators, maintenance rule monitoring guidelines, and the WANO performance indicators. The differences between reporting rules for the three different systems can lead to unintended errors since plant personnel are burdened with the similar yet different rules. Finally, as stated previously, there needs to a formal process established to change PIs or to introduce different ones.

ENCLOSURE 2

NEI Comments On Performance Indicators

The pilot program has provided an excellent opportunity to test the new performance indicators. The pilot plants were able to report data with sufficient accuracy that NRC was able to fulfill its oversight responsibilities. For the most part, the indicators were collected, verified and reported with minor difficulty. The indicators were reported in a timely manner and were used by the NRC to assess performance as planned. The experience and lessons learned during the pilot program will be of great value to the remaining plants in accurately reporting data.

Reporting Difficulties during the Pilot which may be experienced by industry

Pilot plants experienced the most difficulty in reporting data for the safety system unavailabilities; safety system functional failures; ERO drill performance and participation; and security performance index. We believe the great majority of problems have been addressed through bi-weekly public meetings between the NRC and stakeholders. These clarifications and corrections were incorporated in the revisions to NEI 99-02, Regulatory Assessment Performance Indicator Guideline.

Need for formal process to resolve PI interpretation Issues during full implementation

When all 103 plants begin formally reporting data to the NRC, there will be additional questions that will need to be resolved (for example, plants with unique configurations or systems). NEI strongly recommends that the bi-weekly meeting process be continued in order to address these questions in an open and efficient manner.

Security Equipment Performance Index

In most cases historical data was available to assess the indicators to ensure that the data could be collected and applied in assessing performance. The performance indicator thresholds are appropriately based on historical data and the pilot plant program, with one exception, the Security Equipment Performance Index. The Green-White threshold for this indicator is overly conservative and is not based on historical performance. The threshold for this indicator, as calculated, would require an availability for individual security equipment which exceeds that required for the emergency diesel generators and other safety systems, even though the unavailability is fully compensated by the guard force. The White-Yellow threshold for this indicator is inappropriate. The Yellow band for the performance indicators is meant to represent a significant reduction in safety margin. Since unavailable security equipment is fully compensated by the guard force, a significant reduction in margin cannot occur. Note that several other PIs do not have Yellow or Red bands because the safety significance of the indicators cannot be established consistent with other indicators in the program. (for example, unplanned power changes, and safety system functional failures). This indicator requires additional analysis and correction prior to full implementation of the program.

Need for Rigorous change process

The current set of performance indicators in the New NRC Oversight Process were selected on the basis that the information was already being compiled and/or reported and that there was a reporting history associated with the indicators that provided a degree of confidence in the data.
It is recognized that the NRC oversight process is not a static process and that changes, including those to performance indicators, will be a continuing feature of the process.

NEI understands that NRC is already in the process of considering additional performance indicators. The process used to develop the current assessment tools should be applied for any additional ones. That process involved significant effort and analysis. For the PIs, the key steps included: identifying a potential indicator (including determining whether the indicator would provide significant additional risk insights beyond the current indicators or could be used to substitute for current inspection), developing a definition, collecting historical data, establishing thresholds, pilot testing the indicator, and training industry and NRC on the indicator.

This process will involve significant resources and time to ensure that the indicators are effective. We believe, based on experience, that this process to be successful would likely require about 15 months. For example, the following timetable would probably be representative:

Milestone	Schedule
• Review industry operating experience and identify a candidate performance indicator	Jan '00
• Validate that PI addresses the attributes of importance for the appropriate cornerstone	Jan '00
• Obtain stakeholder concurrence on proposed PI	Feb '00
• Develop draft PI definitions and clarifying notes	Mar '00
• Gather best available historical industry data	April '00

• Review historical data and establish tentative regulatory thresholds	May '00
• Pilot proposed PI definitions, thresholds and data reporting at 8 to 10 plants	May'00 to Oct '00
• Revise baseline inspection program to reflect differences in information between old and new indicator	Oct'00
• Evaluate pilot lessons learned and make necessary adjustments	Nov '00
• Train industry on new PIs	Dec '00
• Implement new PI industrywide	Apr '01

Future data reporting and common definitions

The new oversight process and the performance indicators will continue to evolve, as discussed above. Among the improvements that are appropriate to explore are more effective means to report the data and common definitions. Currently certain performance data is provided to WANO. Related and similar data is compiled by utilities to comply with the Maintenance Rule. The performance indicators for the new regulatory oversight process will be sent to NEI for processing, then returned to utilities for review and submittal to the NRC. There are also similar indicators, for example, in the safety system unavailability area, which have three different definitions – one for WANO, another for NRC, and a third for the Maintenance Rule. Efforts should be made to develop common definitions wherever possible.

Additional areas for change are the scram indicators under the initiating events cornerstone. The scram indicators were chosen because they are currently tracked and reported by industry. However, there is some concern that tracking these indicators in the oversight process may provide a negative incentive to manually scram the reactor to avoid "tripping" an indicator. This concern can best be remedied by tracking the frequency of initiating events directly, rather than on the way the plant deals with the event (i.e., a scram). Replacement of the scram indicators should follow the change process described above.

NRC treatment of historical data

The historical data submitted by all of industry, now scheduled for January 21, 2000, will of necessity be a "best faith" effort. This is because many of the indicators were either not in existence prior to the pilot, or have had the definition significantly changed. It is important for NRC to recognize this difficulty, both in its enforcement policy, and in inspections of historical PI data. Errors in reporting data should not be subject to enforcement action. In addition, inspection of historical data should not be so time-consuming for licensees and NRC inspectors that the value of the PIs (to indicate areas where additional attention is needed) is lost. NEI 99-02 provides some agreed upon ground rules for reporting of historical data.

Performance Indicator Results for Annual Assessment

It is NEI's understanding that NRC will use the inspection finding results for the previous four quarters in its annual assessment of plant performance. This is appropriate because the complete set of baseline inspections will be conducted on an annual basis. Performance indicators, however, are revised quarterly to reflect performance over the previous year, or several years. Therefore, it is most appropriate to consider only the most recent quarterly PI results in the annual assessment.

Treatment of historical white inputs when program starts

The purpose of the white band in the performance indicators is to provide an indication that performance has degraded somewhat and additional licensee attention is needed and that NRC should review that action. When the new regulatory oversight process begins, there may be some plants with white performance indicators that do not reflect current weaknesses, and thus do not need a supplemental inspection. For example, there may be safety system unavailability results that reflect a problem that has been identified, corrected and reviewed by NRC in the past; there is obviously no need to follow up again. Some of the indicators were not in existence in the past and are not actually regulatory requirements. Therefore licensees did not focus management resources in these areas in the past. An example is the ERO participation PI. It would be unreasonable to conduct additional supplemental inspections in this area for historical data.

ENCLOSURE 3

Comments On Inspection Process for the Pilot Program For The New Regulatory Oversight Program

General Comments

Overall, the conduct of inspections during the pilot program was good. In general, the inspectors were well prepared, followed the scope and attributes listed in the inspection procedure, focused on risk significant issues, and maintained an open dialogue with licensee staff so that issues could be understood and addressed. An exception was the fire protection inspections in which there was very little dialogue. This is a completely new inspection procedure which will require significant effort prior to full industry implementation.

The amount of time to conduct an inspection varied depending on what was found during the inspection and the training level and experience of the inspector. Work is needed to ensure that the procedures are consistently applied across regions and within regions. This is a matter of training and communication -- and leadership at the national level -- to ensure that regional or individual inspector peculiarities are managed to ensure that the focus is maintained on safety and risk-informed issues. The procedures should be evolutionary, reflecting a learning organization.

The actual amount of time devoted to inspection during the pilot varied considerably from plant to plant and compared to the initial estimates. This is not to be unexpected in a pilot program testing new procedures. In some cases, it appeared that the inspections could have been completed in less time, in other cases more time appeared to be necessary. Based on the pilot plants' experience with the baseline inspection program, NRC management must continue to establish challenging targets for inspection resource requirements. It must also provide sufficient monitoring of inspection activities to ensure that resources are used efficiently.

NEI and the pilot plants have found the new regulatory oversight processes to be more effective. This conclusion is based on the fact that the inspectors now have inspection modules which are more focused on safety issues and equipment, and the assessment process itself encourages inspectors to devote their efforts to potential problems which are more likely to be of safety significance.

By increasing the focus on risk significant attributes, as opposed to procedural compliance (particularly licensee compliance with its own procedures, not NRC regulations) the NRC can reduce its resource load and at the same time provide greater safety assurance. By focusing on results and risk rather than blanket coverage, the NRC can be more resource efficient without decreasing its assurance of safety. Compared to the previous core inspection program, it would seem that the baseline inspection would not necessarily involve fewer inspection hours; however, for a plant with mostly all green findings, there should be less need for region based inspection. Hence the total inspection effort should be less.

The inspection procedures themselves address the cornerstones of the program. They are written in a clear manner that should be capable of being consistently applied across regions and plants. The problem identification and resolution inspection procedure contains procedural detail that seems at odds with a results oriented inspection. Initial inspections in this area appeared too process oriented; the more recent ones seem to have shifted the focus more appropriately to results.

A major improvement in NRC inspection efficiency and effectiveness could be achieved by relying more on licensee self-assessment activities. If the licensee conducts a self-assessment that addresses the same attributes as an NRC inspection module, NRC should be able to reduce its total inspection effort. For example, the NRC could review the scope and results of the licensee self-assessment, and if adequate, need not expend its own resources by redundantly reviewing the same area.

NEI would suggest that NRC consider the following changes to the baseline inspection program to further improve efficiency:

- The frequency of the permanent plant modification module be revised to match the plant refueling outage schedule. It is unlikely that significant modifications will be made to risk-significant systems outside of refueling outages. The current annual frequency is unnecessary.
- The problem identification and resolution inspection can be adjusted to a biennial basis after a satisfactory baseline inspection. Reasonable assurance of the corrective action program integrity will be provided by other inspections that, by design, test on a continuing basis the effectiveness of the corrective action program. The module should include a provision that the frequency can be readjusted to yearly if a decline in performance is observed.
- Radiation monitors are inspected in the radiological controls modules and the maintenance rule modules. The scope of these two efforts should be coordinated to avoid redundancy. In addition, the safety-risk basis for the inspection efforts in each are should be made clear.
- The permanent plant modification module and the 50.59 and SAR change modules could inadvertently review the same design change

packages unless these modules are coordinated. NRC should review the module scopes and the planning practices to ensure that there is no redundancy between these two inspection efforts.

- The permanent plant modification module and the safety system design and performance module can also review the same areas. NRC should review the module scopes and the planning practices to ensure that there is no redundancy between these two inspection efforts.

NEI believes that the inspection planning process can be performed in an effective manner to support the assessment cycle. While inspections were rescheduled more frequently than expected due to the nature of the pilot program, it was clear that the workload is manageable. The structured approach outlined in the action matrix will help the planning process by eliminating many of the unnecessary reactive inspections that have been conducted in the past. In addition, for white findings, the action matrix sets an expected time frame for the follow-up inspections (after utility corrective action activities are well underway) that will facilitate better planing and more effective inspection activities.

Generally speaking, licensee resources for collecting pre-inspection data and for responding to non-significant issues has been less. There has been an increase during the pilot of resources required for general inspection support. NEI and the pilots believe this is due to starting up a new process (for both licensees and inspectors), necessary changes to clarify the PIs, and the compression of the inspection program in order to test a year's worth of inspections during the pilot. Efficiencies in PI collection and in conducting inspections will be gained with experience. Overall, there will be gains in effectiveness and efficiency with an increased focus on safety issues.

In general, the industry has found that the scope and frequencies of the baseline inspection procedures are adequate to address their intended cornerstones. A possible exception is the Fire Protection inspection module and its SDP. At this time, no fire protection inspection has been completed – two are underway – and so no conclusion regarding the effectiveness of this procedure can yet be made.

There are several areas in which improvement to the program is possible:

- Inspection procedures were not updated in a controlled manner, in that inspectors were sometimes using different versions of procedures which had not been provided to the pilot plants.
- In general, the reports do not appear to always closely follow the NRC Inspection Manual Chapter 610*. It appears that the examples in the exhibits of the Manual Chapter 0610 do not reflect the instructions provided in the procedure.

- The identification of a findings, URIs or NCVs in reports is not consistently or uniquely identified in the report. Additionally, they are not consistently reported in all sections of the report
- The Cover Letters are not consistent in their presentations
- Report length and write-up are not always consistent with the new Report Format procedure.
- Quality/Attention to detail of the reports varied and reflect the newness of the Manual Chapter 0610*.
- Some instances were identified where inspector opinions were inappropriately included in the reports.

NRC Inspection Manual, *Manual Chapter 0610 (May 19, 1999 revision)**

<u>Section 0610-03</u>

- The definitions of findings and observations should be improved to reflect the intent of screening criteria and significance determination process.

<u>Section 0610-05</u>

- The discussion of treatment of observations in paragraph 05.01 should be revised to reflect the minimum threshold flowchart.
- The guidance on minor violations in paragraph 05.02 should be updated to reflect the September 29, 1999 guidance memorandum from the Office of Enforcement.
- The treatment of non-enforcement related issues in paragraph 05.02 should be updated to reflect the screening criteria included in the minimum threshold flowchart.
- The information on reporting writing in paragraph 05.03 should refer to the plain language principles for consistency.
- The information on the treatment of significant findings in paragraph 05.04 should be expanded to identify when the additional utility information will be requested and the method of request.
- The information in paragraph 05.05 should be revised to reflect lessons learned on the treatment of external event issues that comply with the licensing basis for the plant. In particular, the guidance should provide cautions on the proper use of assumptions on initiating event frequency changes.

Section 0610-06

- The information in paragraph 06.01 should be revised to reflect the guidance on letter writing in September 20, 1999 guidance memorandum from the Office of Enforcement (EGM 99-006).
- The information in paragraph 06.03 should be revised to reflect the guidance on report writing in September 20, 1999 guidance memorandum from the Office of Enforcement (EGM 99-006).
- The guidance on PIM entries in paragraph 06.03 should note that the affected unit)s) should be clearly identified to ensure proper identification on the unit-specific web pages.

Section 0610-07

- The information on reporting writing in paragraph 07.01 should refer to the plain language principles for consistency.

Inspection Manual Procedures

NEI is providing the following comments on the draft inspection manuals provided during the public meeting held on March 24, 1999. The bold and underlined headings identify the specific inspection manual. The focus area of the inspection manual comments is provided with shading for ease of reference. The NEI comments are provided in italics below the referenced inspection manual text.

PERFORMANCE INDICATOR VERIFICATION

Specific Guidance

03.01 **Each indicator is listed below with the definition from the PI Reporting Manual** and guidance on the verification of the data. Additional clarification of the PI definitions and examples are provided in the PI Reporting Manual. Table 1 provides additional verification guidance by listing the reported elements of each PI and suggesting records for the inspector to review.

NEI Comment: It is ill advised to try to keep this current with PI Guideline. It is better to reference the PI Guideline.

b. **Risk-significant** Scrams

Verification: Perform verification at the same time as the Unplanned Scrams per 7000 Critical Hours. Review licensee's basis for including or excluding each scram in the **Risk-significant** Scrams PI. Compare licensee's basis with data from licensee event reports, monthly operating reports, and operating logs to determine if the scram was properly classified.

NEI Comment: Change description of indicator to match guideline.

e. Safety System Failures

Definition: The number of actual or potential failures of the safety function of the monitored SSCs in the previous 12 months. The following SSCs are monitored:

The threshold for reporting this performance indicator is that the system failure is reportable under 10 CFR 50.73.

NEI Comment: Relocate this sentence to match guideline.

h. Containment Leakage

Definition: The estimated "as found" **integrated** leak rate for the containment as a fraction of the design basis leak rate (L_a).

Verification: Review **integrated** leak rate and local leak rate test results and compare to the design basis leak rate from the FSAR or other design basis documents.

NEI Comment: Change description of indicator to match guideline.

j. Emergency Response Organization Readiness (ERO)

Definition: The percentage of **ERO and operating shift crews** that have participated in a drill, exercise, or actual event in the previous 24 months.

NEI Comment: Change description of indicator to match guideline.

m. Process Effluent Radiological Occurrences

Definition: The total number of process effluent radiological occurrences in the previous **36 months**.

Verification: Review licensee's problem identification and resolution database for liquid or gaseous effluent releases that were reported to the

NRC. For the past **36 months**, ensure that all were counted as PIs. **During plant status reviews, screen plant incidents involving leaking pipes involving radioactive liquids or gases that are not bounded by plant collection systems.**

NEI Comment: Change description of indicator to match guideline. Also last sentence needs work to clarify intent. This should not include annual reporting of normal effluent releases.

n. Protected Area Security Equipment Performance

Definition: **The percent of the time during the previous 12 months that all components (barriers, alarms, and assessment aids) are available and capable of performing their intended function.**

NEI Comment: Change title and description of indicator to match guideline.

o. Vital Area Security Equipment Performance

NEI Comment: Delete section.

Table 1 - Performance Indicator Verification Schedule

Risk-Significant Scrams

NEI Comment: Change description of indicator to match guideline.

Vital Area Security Equipment Performance

NEI Comment: Delete section.

Table 2 - Performance Indicator Verification Inspection Guidance

Risk-significant scrams - *NEI Comment: Change description to match guideline.*

Number of **risk-significant** scrams - *NEI Comment: Change description to match guideline.*

Maximum RCS unidentified leakage - *NEI Comment: Delete.*

Integrated leak rate test results - *NEI Comment: Delete.*

Protected Area Security Equipment Performance - *NEI Comment: Change title to match guideline.*

Vital Area Security Equipment Performance - *NEI Comment: Delete.*

IDENTIFICATION AND RESOLUTION OF PROBLEMS

02.02 Problem Identification and Resolution In Other Inspectable Areas

b. Incorporate checks of problem identification and resolution in ongoing inspection activities for other inspectable areas for all cornerstones. Inspect activities, interview staff, and review documents to determine if the following activities are being conducted in an effective and timely manner commensurate with their importance to safety and risk:

1. Identification of root cause (Appendix B) or contributing causes (Plant Support areas).

NEI Comment: This section needs to be qualified to take into account the fact that Appendix B Criterion XVI only requires root cause for significant problems. Also use of "plant support" terminology is confusing because of PPR/SALP usage of Plant Support area of activity.

XXXXX-03 INSPECTION GUIDANCE

General Guidance

Sampling performed during this inspection should verify that: (1) the licensee's assessments of problems and issues were of sufficient scope to address the key attributes of the cornerstone; (2) the risk-significance of the findings was assessed; (3) root cause analyses and corrective actions were timely and adequate to prevent recurrence; (4) industry and NRC generic issues were considered; (5) recurring issues were identified, and (6) required reports to the Commission were made.

NEI Comment: This inspection expectation establishes requirement for risk analysis that doesn't currently exist. This is an example of regulation by inspection.

Many inspection findings from other inspectable areas may have little impact on the cornerstones when taken individually. However, when the corrective actions for such findings are looked at collectively, potentially risk-significant issues may emerge. Therefore, the inspector should not necessarily approach this inspection solely based on individual system or cornerstone importance. The potential impact of the root cause (Appendix B) or contributing causes (Plant Support areas) and collective impact of the corrective action backlog on the plant as a whole should be sought. Unidentified or uncorrected root cause or contributing causes and improper identification of significance by the licensee for multiple corrective action items could lead to increasing common cause event rates, human event rates, and to breakdowns in multiple cornerstone areas. For example, an individual procedural adherence problem may have no risk significance. However, multiple procedural adherence problems in the Occupational Exposure, Initiating Events and Barrier Integrity cornerstones could.

NEI Comment: NRC should avoid too much subjective judgment here. Inspectors should not be expected to stretch issues and assumptions to "make" *potential risk-significant issues. Similarly, they should not string together diverse problems just to "make" a possible common cause argument. Issues must have real risk-significance in accordance with the defined criteria. There also has to be a clear and logical link between collective problems and a real common cause issue. If this is clearly spelled out, too much speculation on the implications of various findings will divert attention and squander utility and NRC resources on academic debates.*

Item 01.01, page 1 - The stated objective is the wrong objective. Rather than assessing the program, the NRC should be assessing the results of the program. The current objective does not adequately reflect performance based philosophy.

Item 01.02, page 1 - same comment as above.

02.03d, page 3-4 - The description of the interview process to determine a "chilling environment" does not seem very appropriate. NRC management needs to be careful not to encourage a technical inspector to go off on an investigation of a perceived problem in this area without careful guidance and supervision.

03.03a, page 7 - This section should explicitly state that the NRC, in assessing "overall licensee performance" in this area, should focus on the licensee's performance indicators of actual results of the CAP.

03.03a, page 8 - This section directs the inspector should sample NCVs in all the cornerstone areas. This should already have been done for the inspector in other inspections or in reviewing the PIM. It appears that the next page of the module makes just this point.

Page 10 - The guidance for looking at an OE should focus on indications that there is a problem in this area. A look at any specific area within the CAP should only be triggered by indications of a problem. Again, the module seems to slip from wanting to be performance based back to looking at the perceived quality of the individual pieces.

The direction to look at a SRAB's trending programs implies that there are separate one for their use. This whole section needs to be rethought to focus on results rather than component pieces.

Adverse Weather Preparations

02.01 Work Planning

Using plant-specific data, identify which external weather sources have the highest risk ranking(s) and the times of the year when this risk is highest. **Identify which safety-significant SSCs are potentially the most affected by the external weather. Using the site specific risk information and Appendix A, identify one nonfailure tolerant SSC and one site-specific high-risk ranking SSC for review.** For purposes of this inspection, a nonfailure tolerant SSC shall be defined as a component or system which is not a major contributor to plant risk (low contributor to core damage frequency) because it is highly reliable, but if it were to fail it would cause high risk (high CCDP). Prior to entering the time of the year when the risk is highest and/or when information gathered during plant status reviews indicates impending severe weather, review the licensee's preparations for addressing that risk.

NEI Comment: Need to be careful that inspectors do not transfer the burden of selection of risk-significant inspection areas to the licensee.

Changes to License Conditions and Safety Analysis Reports

Attachment A - Action matrix

Security Plan - **10 CFR 50.90 Application for Amendment** Was Required

NEI Comment: It is unclear how 10 CFR 50.90 applies when 10 CFR 50.54(p) is applicable.

Emergent Work

APPENDIX A: INSPECTION GUIDANCE

Mitigating Systems (60%) - Risk Priority

Emergent work when high risk configurations already exist, due to planned on-line maintenance.

NEI Comment: The matrix entry can be interpreted that these issues are additive. It should be clarified that new emergent work must have some clear link back to a cornerstone objective to be an impact. Similarly, multiple issues that have risk impact need to be linked in risk-space to be considered additive. If we are not clear on expectations, any emergent job during a period may become the focus on increased inspection regardless of its nexus to safety. Similarly, two simultaneous risk-significant configurations for diverse events may become the focus when the actual safety linkage does not exist.

Equipment Alignment

LEVEL OF EFFORT: Periodic control room equipment alignment checks. Partial system walkdowns approximately monthly to verify operability of redundant train/system with other train/system inoperable or out-of-service. **One complete risk-important system walkdown approximately every 6 months.**

NEI Comment: Is this a single or multiple unit requirement?

Event Follow-up

02.03 Confirm that the licensee has properly classified the event in accordance with the Emergency Plan and made timely off-site notification of the event when required. If necessary, has the licensee notified the NRC of any new developments or significant changes in plant conditions? Determine if the licensee has identified the root cause of the event and implemented appropriate corrective action before starting the reactor.

NEI Comment: This wording implies all "events" must have a formal root cause analysis. Some events may have obvious causes that require no formal root cause analysis.

Fire Protection

General

This inspection module replaces the following Inspection Procedures: 64150 "Triennial Postfire Safe Shutdown Capability Re-verification", 64704 "Fire Protection Program", 64100 "Postfire safe shutdown, emergency lighting and oil collection capability at operating and near-term operating reactor facilities". One major difference noted is that previously, module 64100 was to be performed once at each plant, and only re-performed at the Region's discretion, based on the findings of the Triennial performance of 64150. 64150 itself focuses on the plant's configuration control process for maintaining the NRC-approved safe shutdown analysis, and does not attempt to question the underlying safe shutdown analysis itself.

New module 71111.05 gives the NRC the direction to go into much greater depth of review in both fire protection and safe shutdown. In the past, if IP 64100 were to be performed after initial plant licensing, several steps are specifically not performed, related to housekeeping, hot work controls, fire risk management, etc., which are now included in the scope of routine inspection under 71111.05. The net result is more frequent performance of the "soup to nuts" style inspections experienced by Salem and Harris (1 week every 3 years).

The compressed schedule of these inspections has shown them to be particularly burdensome to both the NRC and the licensees (historically, large portions of initial licensing 64100 inspections were performed offsite by NRC contractors, due to the volume of material involved). To achieve this same depth of review in a one week time period, in "real time", is not practical, even for a limited number of plant areas. Also, licensees should not have to support large volume document requests during the inspection itself.

The licensing basis of the plant needs to be taken into consideration in regards to NFPA code of record. During the Pilot Plant inspections, the NRC inspectors initially evaluated fire protection features against the current standards to determine the overall effectiveness of the system. However, when a potential inspection finding is identified, the NFPA code of record should be used to evaluate the system. The September 13, 1999 draft does not specifically address NFPA codes.

The amount of material that is necessary to complete the scope of the triennial inspection is enormous. In order to make the inspection closer to a one week inspection, the NRC needs more time between the information gathering trip and the actual inspection. After the information gathering trip, there should be an additional meeting with the licensee to discuss the safe shutdown analysis for the particular plant so the NRC can refine the material needed for the inspection and the Licensee can gather this information prior to the on-site inspection week.

The inspection guidance covers the programmatic requirements of ensuring post-fire safe shutdown in accordance with the applicable regulatory requirements for that facility (10CFR50.48, Appendix R, etc.). However, strict compliance with the programatic requirements does not necessarily translate into risk informed findings. For example, cable tray fire wrap may be identified as an inspection finding since it does not meet the one-hour requirement of Appendix R. However, the fire wrap from a performance basis may have adequate fire resistance that is greater than the hazards of the room. Therefore, there may be little or no risk impact to not meeting the licensing basis requirement. The assessment of degradation is very subjective and can vary significantly from inspector to inspector.

Additonally, the inspection guidance does not select the inspection activties to occur in the fire areas of greatest risk significance.

Comments on Resident's Routine Monthly Inspection:

In IP 71111.05, the general philosophy of the resident's inspection objective is described as:

> "The resident inspector should not attempt to address all plant areas each month. The monthly plant tour should focus on from two to four plant areas important to risk. The resident inspector should note transient combustibles and ignition sources (and compare these with the limits provided in licensee administrative procedures). The resident inspector should also note the material condition and operational status (rather than on the design) of fire detection and suppression systems, and fire barriers used to prevent fire damage or fire propagation."

As described above, the Resident is not expected to focus on the actual design of the fire protection features, but rather on the material condition and operational status. Contrary to this objective, several inspection topic areas expect the Resident to make some form of judgement regarding the design of fire protection features:

- "Verify that the fire detectors installed in the room are located near or on the ceiling."
- "Observe that sprinkler heads are located near the ceiling and under major overhead equipment obstructions (e.g., ventilation ducts)."
- "Observe that the gaseous suppression system (e.g. Halon or CO2) nozzles are located near the ceiling and are not obstructed or blocked by plant equipment. ... Observe and verify that the dampers/doors will close automatically (or their closure is otherwise assured) upon actuation of the gaseous system."
- "Ensure that adequate numbers and types of portable fire extinguishes are provided at designated places in or near the area being inspected, and that access to the fire extinguishers is unobstructed by plant equipment or other work related activities."
- "Observe and verify that a hose station can provide coverage for the area being inspected (maximum hose length 100 feet and an electrically safe fog nozzle). Observe and verify that the water supply control valves to the standpipe system are open and that the fire water supply and pumping capability is operable and capable of supplying the water flow and pressure demand."

- "Observe the condition of the fire dampers in the areas being inspected. Ensure fusible link fire dampers are not prematurely shut or obstructed." *This is especially difficult since dampers are inside HVAC ductwork and require disassembly for inspection.*

These inspection items in particular, require the resident to make judgements regarding the design of various fire protection features. These items are not subject to frequent plant changes, and would reasonably be expected to remain as they were originally installed (e.g., detectors installed on the ceiling will most likely remain on the ceiling). Furthermore, the importance of these design details on overall fire risk does not rationalize the frequency of their observation. For example, whether a detector is installed on the ceiling, or below it may change the response time of the detector, but will not completely prevent it from working, whereas a detector that is out of service would most likely not work.

These design-related judgements may be difficult for the resident to readily make (remember, 1 hour a month to do this), and the added vigilance will not lead to a dramatic fire risk reduction. These types of inspection details were formerly included in IP 64704, which was performed annually, with the assistance of a Region FP specialist. It is recommended that these specific aspects be returned to the annual portion of the inspection process.

To get the most beneficial Fire Risk reduction for the inspection effort expended, the Resident's monthly tours should focus on the effectiveness of administrative controls of plant work activities (fire initiator reduction), housekeeping effectiveness (prevents the growth of incipient fires), and the operability status of FP mitigating systems (detection & suppression systems are highly reliable and effective, unless they are INOP or blocked). These types of inspections can be readily performed, and provide the greatest risk benefit for the amount of inspection effort expended. Typically, "findings" in these areas are readily corrected, resulting in an immediate risk benefit to the plant.

Comments on Resident's Annual Routine Inspection:

IP 71111.05 only contains inspection requirements pertaining to Fire Brigade. The bulleted items from the above discussion of monthly inspections should be relocated to the annual inspection section, or removed entirely. Suggestion for their removal is based on observation of NRC inspection at Salem, where these same topics were reviewed under the guise of the "Triennial" portion of this IP. Their ultimate fate (annual or triennial) should be reflected in the IP.

IP 71111.05 currently does not contain any guidance for the Resident as to how to go about performing an annual inspection. Annual inspection guidance should be developed.

Comments on Region's Triennial Inspection:

General Comments

This portion of the IP appears to be an "FPFI in disguise". As such, it calls into question the original design and licensing basis for the plant, by directing the inspector to "pass judgement" on the acceptability of every aspect of FP and FSSD design. Compliance to NRC requirements are based on the inspector's judgement, and not on the licensing basis. Previously, the triennial component (IP 64150) provided assurance that the Licensee could make plant changes without adversely affecting their FSSD and FP programs. IP 64150 was originally meant to take 5 days to perform, so clearly with the added scope and depth included in 71111.05, 5 days is insufficient.

The general philosophy of the revised inspection process is that poor performing plants receive more NRC scrutiny than well performing plants. The triennial portion of IP 71111.05 in particular is not consistent with this objective, since it prescribes a fixed inspection frequency, as opposed to a "for cause" threshold. It also does not recognize the industry's commitment to strengthen our own self-assessment efforts, in exchange for reduced NRC oversight burden for well-performing plants. Generally speaking, it appears that burden has only been added, without any attempt to reduce burden for plants with effective self assessment processes and performance trends.

SECY 99-140 "Recommendation For Reactor Fire Protection Inspections" states:

> "The staff believes that future fire protection inspections should be more comprehensive and risk informed than the current core inspections. For example, future inspections should address the existing regulatory requirements regarding post-fire safe-shutdown capability, which are not inspected under the current core inspection program, with emphasis on activities, plant areas, and safe-shutdown configurations where the potential risks are greater."

The problem with this opinion from SECY 99-140 is that presumably, the NRC at one time found licensee's Safe Shutdown analyses acceptable. Re-inspection of these analyses is not likely to reduce plant fire risk in a cost effective way, due to the tremendous amount of effort which goes into the creation of these types of analyses, as well as their inspection. Similarly, re-inspection of these analyses on a frequent basis is not likely to result in appreciable safety benefit.[1] Reviewing all existing designs and analyses for risk significance is not the intent of the oversight process. A more cost effective alternative would be to inspect the licensee's change processes which must maintain these analyses as plant changes occur, and to assure a high availability of suppression and detection systems, and strong administrative controls over hot work activities and housekeeping. Presumably, a licensee who makes dramatic changes to their safe shutdown analysis could have those changes inspected under an inspection process that looks at the licensee's change mechanisms, without routinely performing a baseline inspection of their safe shutdown analysis.

One observation made during the November 1999 pilot inspection at Salem was that the NRC reviewed Salem's Fire protection SSCs to the current NFPA Codes, instead of the codes of record. When questioned regarding this practice, the NRC stated that although an SSC may comply with the code of record, if it does not comply with the current code year, it could be considered "Degraded" when it is factored into the SDP. Calling a system degraded when it complies with the licensing and design bases is not appropriate, and a system that complies with the code of record is not necessarily degraded in comparison with a system that complies with the current code version. This appears to be flawed reasoning, since the Fire IPEEE's are based on Nuclear Power Industry Experience for Fire SSC reliability. Any "defects" in older versions of NFPA codes that affect SSC reliability would be implicitly included in the reliability data utilized in the development of the IPEEE's. Considering these SSCs as "degraded" under the SDP amounts to a "double jeopardy". The industry should be very concerned if this new inspection process is being used to backfit new staff positions or code requirements on plants that meet their current design and licensing bases.

Specific Comments

IP 64100 contained separate sections for "Redundant Safe Shutdown" (Appendix R III.G.2) and "Alternative Shutdown" (Appendix R III.G.3 and III.L), since the requirements of each are in some cases dramatically different. IP 71111.05 has combined these two sets of criteria in a way that carries the most restrictive Alternative Shutdown criteria over such that they are now being applied to Redundant Shutdown systems.

Several "Backfit" items were observed in section 02.03 "Triennial Inspection" of IP 71111.05. Unless otherwise noted, these are new additions to the IP, and did not appear in IP 64100 or IP 64150.

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[1] Risk evaluations performed by licensees in response to FSSD LERs and FPFI findings have typically shown that individual non-conformances with FSSD requirements have resulted in very low safety significance.

- 02.03.b.1.b - The requirement to maintain the reactor coolant level above the top of active fuel is specified as a requirement for all "redundant" shutdown strategies. This requirement is taken from 10CFR50 Appendix R, Section III.L, which only applies to "Alternative Shutdown" capability. This has previously been upheld by "Connecticut Light & Power vs. the NRC". This is a change in wording from IP 64100, which specifically permitted the use of low pressure systems and brief core uncovery for BWRs.
- 02.03.b.1.a - states "The reactivity control function is capable of achieving and maintaining cold shutdown reactivity conditions." This is probably not a backfit, as much as a misunderstanding of the role of the reactivity control capabilities of light water reactors. The reactivity control capability, by itself, cannot take a unit to cold shutdown, it can merely establish "sub critical" conditions in the reactor. Note: this is a carry-over of an error from IP 64100.

Minor comment: Section 02.03.b.3 "Post-Fire safe shutdown circuit analysis" is a new addition to the inspection scope. Items (a) "Common power supply/bus concern" and (d) "Fuse/breaker coordination" are redundant. Items (a) also requires that an analysis be performed for "Multiple High Impedance Faults". MHIF has been promulgated indirectly as an "NRC Staff Position" via GL 86-10, however it has never been promulgated in any formal way that requires licensee compliance. Several plants have received their SERs for Appendix R without this requirement. It's inclusion in the inspection procedure may be a "backfit" for individual plants.

Section 02.03.b.5 "Operational Implementation of Alternative Shutdown Capability" is a new section. This section includes verification that Alternative Shutdown procedures exist, sufficient operators are available to perform the shutdown, and performs a procedure walkthrough. This section contains two items that may be "backfits" at some plants:

- "Verify that the licensee has incorporated the operability of alternative shutdown transfer and control functions into the plant technical specifications.
- "Verify that the licensee periodically performs operability testing of the alternative shutdown instrumentation and transfer and control functions. In addition, verify that if the licensee imposes the appropriate compensatory measures during periods in which alternative shutdown capability may be declared inoperable."

The inclusion of alternate shutdown capability within plant's technical specification has been requested by the NRC on a case by case basis, based on NRR's review of licensee's proposals for alternative shutdown modifications.

The scope of SSCs included in each plant's tech specs for Alternative Shutdown can vary widely, and in some cases may have been relocated to other places by NRC approved processes (ex., "Improved Technical Specifications" projects, and relocation of Fire Protection out of Tech Specs per GL 88-12). Similarly, the application of specific compensatory measures when the alternative shutdown capability is declared inoperable may vary widely between plants. If the NRC has a specific expectation about what they wish to see in terms of compensatory measures, they must first promulgate them via the appropriate process.

Section 02.03.b.7 "Emergency Lighting" has been re-written to imply that emergency lighting is only provided for "Alternative Shutdown". Emergency lighting may be relied upon for any shutdown capability, not just alternative. This should be clarified. The wording in this section also implies that emergency lighting will be provided by fixed units; however, some plants have found that 10CFR50 Appendix R does not specifically require that the lights be "fixed", and have shown that they can also comply by providing 8-hour portable lighting (e.g., helmet lights). This is a fairly new concept that has come about due to improvements in the technology of portable lights (and reliability problems with the fixed lights). This flexibility is allowed under the rule, and should be reflected in the IP.

This section also contains expanded requirements from IP 64100 regarding observing manufacturer's testing and maintenance requirements, and defining specific testing protocols (blackout tests). These details should be deleted for the following reasons:

- EPRI has performed a comprehensive review of the reliability of self-contained battery powered emergency lights.[2] For many models of lights, these tests have shown that testing and maintaining the lights per the manufacturer's recommendations does not result in a high reliability. The Industry has been working with the manufacturers and EPRI for several years to develop testing protocols that are more predictive of equipment failure, and reduce maintenance costs. By prescribing specific testing protocols, IP 71111.05 could result in a "step backward" in e-light reliability and maintenance costs. The NRC has already recommended that e-lights be included within the scope of the maintenance rule in Information Notice 97-018. It is recommended that the details of maintenance and testing be left up the licensees. If the NRC is concerned about the reliability of E-lights, this section should verify that licensees have taken action to assure a high reliability for the lights, without

[2] EPRI TR-100249 Rev 1, "Emergency Battery Lighting Unit Maintenance and Application Guide"

prescribing unnecessary methods of doing so.

- This section also contains prescriptive test performance criteria for determining if adequate illumination is installed (e.g., blackout tests). Blackout testing has not been universally accepted as an appropriate testing methodology by all licensees. The success criteria "Determine if illumination is adequate to perform required shutdown actions" is too subjective. NRC guidance and plant specific commitments should be considered in this type of review.

<u>**Comments on Resource Estimates**</u>
IP 71111.05 does not provide a resource estimate for the annual portion of the inspection. The resource estimates provided for the monthly, and triennial components appear to be low. For the triennial component, 4 inspectors on site for one week would result in 160 inspector-hours, not considering inspection preparation, close-out, after-hours activities (observing back shift fire drills) and substantial overtime (observed at Salem inspection Nov 1999).

INSPECTION BASES: **For most reactor plants, fire is the dominant risk contributor to external events (earthquakes, floods, high winds, etc.).** Fire protection defense-in-depth is accomplished through control of combustibles and ignition sources, mitigation of fires that do occur through fire detection and automatic and manual suppression capability, and a well analyzed and implemented post-fire safe shutdown capability. Safe shutdown capability includes the existence of adequate fire barriers to establish the fire area or fire zone configuration and to ensure the shutdown equipment functionality assumed in the post-fire safe shutdown analysis. If defense-in-depth is not maintained through a well functioning licensee fire protection program, post-fire safe shutdown of the plant may be challenged. This inspectable area verifies aspects of the Mitigating Systems cornerstone for which there are no indicators to measure performance.

NEI Comment: What is the basis for this statement? Why is it relevant here? The focus should be on risk-significant elements of a particular plant. It is not effective to create a heightened sense of concern with the inspection force based on some issues from a few plants.

LEVEL OF EFFORT: On a monthly basis, the resident inspector will tour **high fire risk plant areas** to assess control of transient combustibles and ignition sources, fire detection, manual and automatic suppression capabilities, and barriers to fire propagation.

In addition, every three years, an inspection team consisting of a fire protection engineer, mechanical engineer, and electrical engineer will conduct a one week, **risk-focused**, onsite inspection of all three components of defense-in-depth, with major emphasis on post-fire safe shutdown capability and configuration management.

NEI Comment: Is there a clear sense of the standards used to determine this risk-focus? If not, it is essential that agreement is reached. Otherwise, the general concerns from a few plants will be imposed on all plants because nothing better exists. Also, the differences between the protection schemes required by Appendix R and the different scenarios that may be evaluated in a fire risk study must be reconciled. The first set is the requirements whereas the second set is not. Failure to reconcile these differences can result in additional regulation by inspection.

02 INSPECTION REQUIREMENTS

02.01 Routine Inspection. Conduct routine reviews of fire protection program observables. The resident inspector's assessment of the licensee's control of transient combustibles and ignition sources is addressed on a more frequent basis in the Plant Status inspection procedure. **Select high fire risk areas based on the plant-specific risk study or on RIM2.**

a. Monthly the resident inspector will tour **high fire risk plant areas** to assess fire detection and manual and automatic suppression capabilities, barriers to fire propagation, and fire protection-related compensatory measures.

b. Annually, the resident inspector **will observe a fire brigade drill in a high risk fire area**, or actual response of a plant fire brigade in any plant area.

NEI Comment: Is there a clear sense of the standards used to determine this risk-focus? If not, it is essential that agreement is reached. Otherwise, the general concerns from a few plants will be imposed on all plants because nothing better exists. Also, the differences between the protection schemes required by Appendix R and the different scenarios that may be evaluated in a fire risk study must be reconciled. The first set are the requirements whereas the second set is not. Failure to reconcile these differences can result in additional regulation by inspection.

NEI Comment: This section also sets an explicit expectation for a licensee who conducts a fire drill in a "high risk area." There is no requirement in this area. This inspection guidance is overly prescriptive and establishes regulation through inspection.

02.02 Triennial Inspection. Conduct a one-week triennial team inspection of the licensee's fire protection program emphasizing post-fire safe shutdown capability and configuration management.

a. The inspection team leader will manage and coordinate a 2-3 day information gathering site visit accompanied by the team members and the senior reactor analyst (SRA) designated to support the team. The SRA will provide a report to the team leader containing **fire risk results**. The team leader will use the **fire risk results** report and input from the other team members to develop an inspection plan.

b. The team leader will manage and coordinate the conduct of a one-week triennial fire protection inspection emphasizing post-fire safe shutdown capability and configuration management. The inspection should be either plant area-based or system-based. The triennial inspection will also include observation of a simulated post-fire safe shutdown from outside the control room (e.g., from a remote shutdown panel and/or remote control stations), and observation of a fire brigade (and possibly offsite fire department) drill for a simulated fire in a **high risk area**.

NEI Comment: Same comment on availability of, agreement on, and use of fire-related risk information.

INSPECTION GUIDANCE:

General Guidance

Triennial Inspection. The triennial inspection is intended to apply, in an integrated, **risk-focused** and **synergistic manner**, sufficient resources for the potential development of **risk-significant findings**. The inspection will focus on selected plant area post-fire safe shutdown scenarios and/or systems.

NEI Comment: Same comment on availability of, agreement on, and use of fire-related risk information. What criteria will be used to establish the risk-significance of the findings?

The team members should determine the plant's current post-fire safe shutdown licensing basis through review of **NRC safety evaluation reports (SER) on fire protection, the plant's operating license, Updated Final Safety Analysis Report (USAR), and approved exemptions** or deviations.

NEI Comment: NRC should provide their own access to this information since we are required to submit it to them. Licensees receive routine requests to *provide this type of information to inspection teams as a convenience; however, it is an added burden to the industry since it duplicates required submission of information.*

Specific Guidance

03.01 Routine Inspection

a. The Specific Guidance Section and Appendices A and B of the referenced draft FPFI inspection procedure provide guidance for the review of fire detection and manual and automatic suppression capabilities, and barriers to fire propagation. Guidance for the review of compensatory measures is in IN 97-48, "Inadequate or Inappropriate Interim Fire Protection Compensatory Measures."

NEI Comment: This new inspection manual was supposed to replace the FPFI. Based on these references, it appears that it could be the FPFI under a new name and using draft guidance. The reference is inappropriate and should be deleted.

c. The Specific Guidance section and Appendix A of the referenced draft FPFI inspection procedure provide guidance for annual observations of fire brigades.

NEI Comment: This new inspection manual was supposed to replace the FPFI. Based on these references, it appears that it could be the FPFI under a new name and using draft guidance. The reference is inappropriate and should be deleted.

03.02 Triennial Inspection.

The triennial inspection is intended to provide a risk-focused look at all three components of defense-in-depth with major emphasis on post-fire safe shutdown capability and configuration management. The inspection will adopt, as appropriate, techniques developed in the referenced draft FPFI inspection procedure, which was written for the more extensive (two weeks onsite) "full-scope" pilot FPFI inspections. The inspection team should consist of a fire protection engineer, mechanical engineer, and electrical engineer. The risk focus will be provided by a senior reactor analyst who will provide extensive input into the planning process.

NEI Comment: Same comment on availability of, agreement on, and use of fire-related risk information. Same comment on inappropriate use of draft FPFI inspection guidance.

The senior reactor analyst will develop and present to the inspection team leader a fire risk results report using the methodology of Appendix I of the referenced draft FPFI inspection procedure. The report will identify plant-specific, fire risk-significant plant areas, structures, systems, components (SSCs), and operator actions developed from risk information such as Independent Plant Examinations of External Events (IPEEEs). The fire risk results report may also consider:

NEI Comment: Same comment on availability of, agreement on, and use of fire-related risk information. Same comment on inappropriate use of draft FPFI inspection guidance.

The inspection plan for the triennial inspection should take into account:

- the fire risk report,

NEI Comment: Same comment on availability of, agreement on, and use of fire-related risk information.

b. The inspection is planned for a one-week onsite inspection period. The areas or systems selected for review will be determined by the inspection plan. For those features selected in the plan, specific inspection guidance pertaining to the features is contained in the referenced draft FPFI inspection procedure.

NEI Comment: Same comment on inappropriate use of draft FPFI inspection guidance.

The SRA will provide input to the inspection report if significant problems are found regarding the validity and/or completeness of the IPEEE information and the degree to which it is reflected in the licensee's fire protection program and implemented in the licensee's post-fire safe shutdown capability.

NEI Comment: Same comment on availability of, agreement on, and use of fire-related risk information. What criteria will be used to establish the risk-significance of the findings?

The inspection team leader should ensure that, when risk-significant inspection findings are developed in one or two of the three areas of defense-in-depth, adequate inspection effort is applied in the inspection of the other area or areas of defense-in-depth.

NEI Comment: What criteria will be used to establish the risk-significance of the findings?

REFERENCES:

NRC "Fire Protection Functional Inspection (FPFI) Draft for Prairie Island Inspection, April 6, 1998"

NEI Comment: Same comment on inappropriate use of draft FPFI inspection guidance.

ATTACHMENT
ROUTINE INSPECTION GUIDANCE TABLE

CORNERSTONE	RISK PRIORITY	EXAMPLES
INITIATING EVENTS (10)	Equipment or actions that could cause or contribute to initiation of fires **in high fire risk areas or near equipment required for safe shutdown.**	
MITIGATING SYSTEMS (90)	Fire Barriers in **high fire risk areas.** Detection Systems for **high fire risk areas** Automatic suppression systems for **high fire risk areas**	Doors and dampers that prevent the spread of fires to/or between **high fire risk areas** remain in place and are functional. Electrical cable fire wraps and penetration seals that protect the **post-fire safe-shutdown train** are not damaged. Fire detection and alarm system is functional for **high fire risk areas.**

NEI Comment: Same comment on availability of, agreement on, and use of fire-related risk information.

Flood Protection Measures

NEI Comment: The expectations for control of breaches is not discussed. Should it be to avoid conflict later?

Inservice Inspection Activities

02.02 Risk-Focused Scope:

The scope of this inspectable area is limited to the following SSCs:

- piping unisolable from the reactor coolant system
  OR
- piping connecting to the RCS for which failure could result in an interfacing system LOCA

NEI Comment: The inspection guidance should also reflect the use of plant-specific information if the utility is using an approved risk-informed inservice inspection program.

02.03 Refueling Cycle Review:

c. Review a sample of at least three welding activities. Verify that the welding activities and acceptance were performed in accordance with Code requirements.

NEI Comment: There should be more guidance on the focus of the welding sample selection to provide a better link to safety.

-03 INSPECTION GUIDANCE:

Steam Generator Tubes

NEI Comment: The inspection module is silent on expectations for steam generator-related inspections.

Inservice Testing of Pumps and Valves - ASME Section XI

P 02.03 d - This section discusses NRC relief for testing through minimum-flow recirculation paths, however, this should be expanded to clarify that relief is not required when outage testing (extended test interval) is done at higher flow rates.

The 4th paragraph of 03 INSPECTION GUIDANCE mentions the current IST inspection procedure 73756. This leads to some confusion since the new inspection procedure should replace the old one.

Nonroutine Plant Evolutions

LEVEL OF EFFORT: The level of effort for this inspection includes reviews of six occurrences per year of operator performance during off-normal or transient operations, one post-reactor trip review, **and 10 LER reviews focusing on operator performance**. Actual level of effort will depend on the numbers of occurrences of off-normal or transient operations.

NEI Comment: This wording suggests that a quota for operator performance LERs may be expected. It should be reworded to better reflect the actual expectations for station operation (i.e., fewer occurrences of operator performance LERs).

Permanent Plant Modifications

02.02 Select Modifications and Determine Applicable Inspection Activities

Review Type	Frequency	Scope and Focus	Applicable Inspection Activities
Biennial Review	1 per year		

NEI Comment: Seems inconsistent.

02.05 Updating Review:

a. Verify that design basis information documents have either been updated or are in the process of being updated to reflect the modifications. Examples of design basis documents which could be affected by modifications are:

- **PRA models**

NEI Comment: Note that no requirement exists to update the PSA models on any specific schedule. Need to ensure that licensees' schedule flexibility and judgment about required changes is preserved.

Piping System Erosion/Corrosion

INSPECTION BASES: Effective implementation of an erosion/corrosion program is important to minimize the potential for high energy fluid system failures that can result in plant accidents. This inspectable area **verifies the Barrier Integrity** key attribute of the Initiating Events cornerstone for which there are no indicators to measure performance.

-02 INSPECTION REQUIREMENTS:

02.02 Risk-Focused Sample Selection

Select a sample of piping for review once per refueling outage. The sample selection should be piping for which erosion/corrosion (or flow-assisted corrosion) is a credible degradation mechanism and **catastrophic failure would result in a high probability of core damage.** Specifically, the selection criteria for piping are:

- erosion/corrosion (flow-assisted corrosion) is a credible degradation mechanism (carbon steel piping **AND** high flow rate)

AND

- **piping unisolable from the reactor coolant system**

OR

piping connecting to the RCS for which failure could result in an interfacing system LOCA

-03 INSPECTION GUIDANCE:

Cornerstone	Inspection Objective	Risk Priority	Example
Initiating Events	Verify that licensee's erosion/corrosion program is sufficient to prevent accidents due to erosion and corrosion	Piping for which catastrophic failure would result in a high probability of core damage	BWR Feedwater Piping Inside Containment

NEI Comment: This inspection module should be limited to BWRs based on the risk-informed scope and purpose.

Alert and Notification System

INSPECTION REQUIREMENTS:

Initial Implementation

1. Review the licensee's siren system testing procedure and determine compliance with the following requirements from NUREG-0654, Appendix 3:

NEI Comment: Should the requirements be listed in the utility's approved emergency plan rather than a guidance document?

2. Determine if each test, as performed, actually tests the elements of the system necessary for the system to perform its design function, e.g., consider if the silent test verifies the ability of the sirens to receive and process control signals to the extent consistent with the system design and the test being conducted:

   - did the siren receive the signal,
   - did it process the signal,
   - did all expected functions respond to the signal

- and is the test designed to verify the ability of the siren to process control signals.

NEI Comment: This criteria is overly prescriptive in regards to what a "silent test" is capable of verifying. In our FEMA acceptance documents, we define a silent test as a test which tests the radio link to the sirens. The majority of siren systems are not capable of testing the system to the level of detail specified in this criteria. This is beyond regulatory requirements and our commitments to FEMA for this type of test.

3. Determine if a growl test is required by procedures and is conducted after all maintenance that could disable a system function.

NEI Comment: "Growl Tests" are conducted as committed to in the REP. However, it is overly prescriptive to require a "Growl Test" after <u>all</u> maintenance activities. Out testing program is defined and our sirens are sounded monthly. The requirement for growl testing after maintenance activities should be deleted.

<u>Drills, Exercise and Event Evaluation</u>

Biennial Exercise

8. During an exercise . . . should be communicated to the licensee and assessed for significance by NRC.

NEI Comment: Is the standard for significance-related to the description of risk-significant areas listed on the next page? If not what is it? A clear definition of risk-significance is needed for this cornerstone area.

<u>Emergency Action Level Revision Review</u>

LEVEL OF EFFORT: Inspection activities in this area include review and assessment of all changes to the EALs on a biennial basis.

INSPECTION GUIDANCE:

All, (i.e., 100 percent), of technical changes to the EALs must be reviewed and approved by NRC. However, purely administrative changes, such as typos, need not be reviewed in detail, other than to verify that they are, in fact, administrative changes.

4. Review the basis for the determination that there has been no decrease in effectiveness. The basis documents for EALs are generally NUREG-0654 and/or NUMARC/NESP-007 Rev. 2 (to be revised as NEI-99-001 and/or NEI-97-003 Rev. 3). Changes that have been formally approved by NRC or that are consistent with the NRC generic EAL guidance documents are not considered to be a decrease in effectiveness.

REFERENCES:

Emergency Preparedness Position (EPPOS) on Emergency Plan and Implementing Procedure Changes, EPPOS No. 4, Rev 1

NEI Comment: The guidance regarding inspection expectations for EAL changes is confusing and should be clarified. The industry position on EPPOS No. 4, Rev. 1, has already been communicated to NRC. This reference should be deleted.

Emergency Response Organization Augmentation

02.03. Augmentation Backup System

b. Review status of backup ERO augmentation system.

XXXXX-03 INSPECTION GUIDANCE

General Guidance

ERO augmentation tests that require personnel to report to their emergency response duty locations are not mandatory, but do provide a high level of assurance that activation goals can be met. However, other combinations of testing and verification can provide a reasonable level of assurance. Commitments on this subject are contained in the licensee Emergency Plan and may vary between sites.

NEI Comment: This section sets an implicit expectation for augmentation staffing tests. As noted, there is no requirement in this area. This inspection guidance is overly prescriptive and establishes regulation through inspection.

03.03. Augmentation Backup System

If the backup augmentation system has not been tested since the last inspection, review the major elements of the backup system to determine if the elements are up-to-date (e.g., Call trees and Call out telephone lists). Determine by interview or, if necessary, by a special drill, whether personnel required to implement the system are knowledgeable regarding the back up system to augment onsite personnel in a timely manner. Coordinate any special drill with appropriate management.

NEI Comment: This section implies that we are required to conduct periodic tests/drills of our backup augmentation system. There is no requirement in this area. This inspection guidance is overly prescriptive and establishes regulation through inspection. This wording will also drive inspectors to request such a test each inspection interval adding burden to the licensee.

Access Control to Radiologically Significant Areas

(Page 4 –Section 02.02) The detailed inspection requirements pertaining to the review of electronic pocket dosimeter (EPD) alarm setpoints should be deleted. There is no basis for placing such a degree specific emphasis in the inspection requirements on the use of EPDs in high radiation areas. The use of EPDs is only one of several options for controls required by technical specifications. For example, the use of EPDs is not required if surveillance over access and work within a high radiation area is provided by an individual qualified in radiation protection procedures. The numerical criteria included in the inspection requirements have no apparent regulatory or technical basis. Such criteria may be taken to imply a regulatory requirement, when in fact, there is no such requirement. For example, typical wording in technical specifications includes a requirement that the "radiation monitoring device...alarms when a preset integrated dose is received," without reference to numerical criteria or to a requirement for a dose rate alarm.

(Page 4 –Section 02.02) The sentence "Determine whether management and administrative controls are designed to maintain exposures ALARA" should be relocated to Attachment 02, "ALARA Planning and Controls." Further the wording should be revised to be with 10 CFR Part 20, which refers to "procedures and engineered controls based upon sound radiation protection principles."

(Page 4 – Section 02.02) The last paragraph refers to "high risk [emphasis added] airborne areas" as having the "potential for individual worker internal exposures of >30 mrem CEDE (12 DAC-hrs)." To improve clarity and consistency, we suggest that the wording be revised to simply refer to "airborne radioactivity areas as defined in 10 CFR Part 20."

(Page 4 – Section 02.03) The reference to "dose rates >25 R/hr at 30 centimeters" should be deleted. Consistent with our comments on the occupational radiation safety SDP, the criterion of >25 R/hr lacks a firm basis in either historical performance or in implied significance. Further, there are no specific regulatory requirements that include such a criterion. For example, neither 10 CFR Part 20 nor standard technical specifications for light-water reactors contain such a criterion.

(Page 6 – Section 02.04a) The reference to "significant [emphasis added] exposures (>1 person-rem)" should be revised to be consistent with the criteria in SDP for ALARA findings. For example, the screening criteria in the SDP are based on 5 person-rem for PWRs and 10 person-rem for BWRs. Further, this item should be addressed in Attachment 02, "ALARA Planning and Controls."

(Page 6 – Section 03.02) The purpose of the discussion in this section is not clear. Further, discussion of inspection of the use of continuous airborne monitors is more appropriately included in Attachment 03, "Radiation Monitoring Instrumentation." It should either be revised to improve clarity and relocated to Attachment 03 or it should be deleted.

ALARA Planning and Controls

General Comments: This inspection procedure should be substantially revised to be more in line with "risk-informed and performance-based" principles as reflected the most current version of the SDP for ALARA findings. Prescriptive detail that goes beyond regulatory requirements should be deleted. The scope and extent of the procedure should be substantially reduced to be more in line with the concept of a "baseline" inspection program and to more appropriately reflect contemporary industry performance.

Suggestions follow for revising this procedure to reflect the general comments (above).

The inspection requirements and guidance should be focused on issues that are relevant to the SDP, i.e., which have a potential to lead to a finding. The SDP addresses ALARA planning and controls for jobs that exceed the criteria specified in the SDP and consideration of overall collective dose as compared with the benchmarks contained in the SDP. Suggested changes to better focus the inspection procedure include the following:

- The sections on source term reduction and control (02.01c and d, 02.03, and 03.03) should be deleted because they are not relevant to the SDP, nor are they performance-based. In addition to being extraneous to the SDP, this material implies requirements that are outside the scope of applicable rules (e.g., 10 CFR Part 20). Such material may be more appropriate for a technical report, e.g., a NUREG.

- The criteria for selecting jobs for inspection should be revised to reflect the criteria in the SDP. For example, Section 02.05 refers to jobs with actual doses greater then 1 rem (versus 5 or 10 rem in the SDP) and Section 02.08 refers to jobs where the actual dose is >1.25 times the exposure estimate (versus 1.5 times in the SDP).

- The section on respiratory protection (02.09) is not an ALARA issue and should be relocated to inspection procedures for emergency preparedness. The inspection objective (01.01) clarifies that this inspection procedure covers "protection of worker health and safety from exposure to radiation from radioactive material during routine [emphasis added] civilian nuclear reactor operation."

- The section on declared pregnant workers (02.07) is not an ALARA issue and should be relocated to Attachment 01, "Access to Radiologically Significant Areas."

- The section on inspecting radiation worker performance (02.04) is redundant to a similar section in Attachment 01 (02.05). It should be consolidated within Attachment 01.

(Page 8 – Section 02.02a.1) The statement that "dose rate gradients (greater than a factor of 2) are often indicative of sources that are not effectively shielded" should be deleted. This statement does not reflect industry experience and is contrary to the concept of "as low as reasonably [emphasis added] achievable" (ALARA). First, without any context, the statement can lead to inappropriate conclusions, for example, the conclusion that an area with a dose gradient of 1 to 3 mrem per hour (i.e., greater than a factor of 2) is "not effectively shielded." Second, effective shielding is the result of an analysis that includes consideration of costs versus benefits in which the benefits are in terms of reduced dose, not in terms of uniform dose fields (i.e., with dose rate gradients less than or equal to 2). Finally, the prescribed use of numerical criteria without reference to specific regulatory requirements is not appropriate for an inspection procedure.

(Page 9 – Section 02.02b) The concept of rotating workers to balance exposures should be deleted because it lacks a regulatory basis and may be contrary to the ALARA principle. Selection of workers to perform specific tasks takes into account a number of factors, only one of which is dose. For example, such factors include needed level of job skills and experience, familiarity with the task, shift schedules, consideration of other tasks needing to be performed, etc. Further, attempts to distribute dose evenly among a number of workers can lead to an increase in the overall collective dose due to variability in specific job skills and experience and inefficiencies associated with work crew changes, shift turnovers, etc. Such a result is contrary to the ALARA principle.

(Page 10 – Section 02.05) The structure of this section implies that there are likely to be multiple occurrences at plants of jobs where actual exposure was more than 50% greater than estimated. For example, the inspection procedure refers to selecting "about 5 jobs of highest exposure significance where actual exposure was greater than estimated by 50%. Industry experience indicates that such occurrences at a facility are less frequent (e.g., fewer than two per assessment period). This section should be scaled down and revised to reflect contemporary industry performance and be more in line with the concept of a "baseline" inspection procedure.

Radiation Monitoring Instrumentation

(Page 16 – Section 02.02) The meaning of the reference to "continuous air monitors associated with the potential for 100 mrem CEDE (40 DAC-hrs)" should be clarified. We suggest wording such as "continuous air monitors used for monitoring airborne radioactivity areas."

Gaseous and Liquid Effluent Treatment Systems

No comments.

Radioactive Material Processing and Shipping

(Page 5 – Section 01.02) – The sentence that refers to the Final Rule on Radiological Criteria for License Termination should be deleted as an inspection objective because it is not applicable to an operating reactor. See further comments regarding Section 02.06b, below.

(Page 6 – Section 02.06b) This section should be deleted because it utilizes inappropriate reference values for detection sensitivity of contamination monitoring instruments. The reference values shown in the table in this section are intended to serve as screening values for surface contamination on building surfaces at the time of license termination. These values are derived from computer models that include a number of conservative assumptions, e.g., the area of surface contamination on floors and walls, re-suspension of the materials, and the presumed annual occupancy in the building. None of these assumptions are relevant to the vast majority of operational situations within the scope of this inspection procedure, nor are these values intended, even in the context of license termination, to be utilized as instrument detection criteria. In addition, the values shown reflect an implied dose of less than 25 mrem, which is not consistent with the SDP that utilizes a dose criterion of 5 mrem.

(Page 8 –Section 03.06) The reference to release of material from the Radiologically Controlled Area (RCA) should be changed to reflect release of material outside to an unrestricted area, i.e., outside of the protected area. The Inspection Objective states that the procedure applies to "...exposure to radioactive material released into the public domain." The boundary of the protected area, rather than the radiologically controlled area, better defines where the public may have limited access. Members of the public do not have unrestricted access within the protected area and are very unlikely to be able to receive exposure from "released" materials. Setting the reference in this section to the protected area boundary still retains a "buffer" on the concept of "public domain" because even outside of the protected area there is some degree of restriction on public access, i.e., within the owner-controlled area.

Radiological Environmental Monitoring Program

(Page 10 – Section 02.02 i) The reference to "overall effect on licensee dose projections" should be deleted. Dose projections are made in accordance with the methodology in the ODCM utilizing effluent sample and monitoring data. Environmental monitoring sample data are not typically utilized in making dose projections.

Access Authorization (AA) Program (Behavior Observation only)

INSPECTION REQUIREMENTS:

a. Inspection Planning:

During review of the **semi-annual** fitness for duty reports, note the number of tests for cause and number of confirmed positives during random testing. If there were **a number** of positive test results in the random testing and no individuals were identified by supervisors to be tested for cause, the inspector should concentrate on this area when interviewing supervisors/managers to determine their understanding of the behavior observation program and **are they having daily contact** or more frequent contact with individuals they are supervising.

NEI Comment: "Semi-annual" reports are expected become "annual" in the new FFD rule—to avoid being incorrect use "NRC-published." The criterion expressed as "a number" is undefined. One is a number. The inspection expectation as stated implies that daily contact is a requirement. There is no *requirement for this. This inspection guidance is overly prescriptive and establishes regulation through inspection. Application of Behavioral Observation program principals on what is sufficient and necessary is a more appropriate discussion to determine how the supervisor does this aspect of his job. There may be other goals of the NRC evaluation that, if intended, should be listed here.*

Determine the number of supervisors, managers and escorts to interview. The number selected should not be less than a total of five supervisors/manager's including contractors and five designated escorts. A minimum of three different disciplines (i.e., Operations, Maintenance, Radiation Protection) should be selected. **The sample should be targeted toward those supervisors who have not identified any member of their staff to be tested for cause**. The inspector should vary the selection of supervisors by group each year.

NEI Comment: This wording suggests that a quota for "for cause" may be expected. It should be reworded to better reflect the actual expectations for station operation. For example, use the last NRC-published data to determine where the station stands relative to the industry. IN 98-39 of October 30, 1998 data for 1997 shows that about 200 of the 94,862 tests for persons covered by NPP FFD programs were required to undergo for-cause tests for observed behavior.

c. Identification and Resolution of Problems

(2) the **risk-significance** of the findings were properly addressed.

NEI Comment: Is there a clear standard for significance related to these problems? If not, a clear definition of risk-significance is needed for this inspection area.

(3) identified nonconformances in the program were evaluated against the ability of an insider to successfully commit radiological sabotage resulting in **a danger** to the public's health and safety.

NEI Comment: The use of danger as a criterion is undefined and open to interpretation. The NRC is in the process of defining the term "radiological sabotage." The standard should be risk-significance. A clear definition of risk-significance and radiological sabotage must be part of any definition needed for this inspection area. The evaluation should be the result of using the agreed upon SDP.

(4) root cause analyses (if required depending on risk-significance) and corrective action were timely and adequate to prevent recurrence.

NEI Comment: Is there a clear standard for significance related to these problems? If not, a clear definition of risk-significance is needed for this inspection area. This is the same as the comment under (2) above.

(7) the performance trend indicated by the sample set was consistent with the applicable PIs.

NEI Comment: It is not clear what is expected here. The expectations should be clarified.

<u>Access Control (Search of Personnel, Packages, and Vehicles: Identification and Authorization)</u>

INSPECTION BASES: Failure of program compromises security barriers in place to protect high risk plant equipment and activities. Risk consequence to radiological sabotage is moderate.

NEI Comment: What is the basis for this statement? Is there a clear standard for significance related to these problems? If not, a clear definition of risk-significance is needed for this inspection area. The criteria should be based on significant (predictable or exploitable) vulnerabilities in the program.

INSPECTION REQUIREMENTS:

c. <u>Identification and Resolution of Problems</u>

The inspector(s) should select a sample set of information comprising, for example, two licensee assessments, three security event reports, one or more audits, three quarters of security drills and exercises conducted with access control equipment operators, and three quarters of maintenance work requests and loggable events. The sample set should cover the period back initially no more than two years. The inspector(s) should use this information to verify that:

NEI Comment: This section will set a specific expectation for the number of self-assessments to be performed. This inspection guidance is overly prescriptive and establishes regulation through inspection. What is the definition of "loggable events?"

(2) the risk-significance of the findings was properly addressed.

NEI Comment: Is there a clear standard for significance related to these problems? If not, a clear definition of risk-significance is needed for this inspection area.

(3) identified nonconformances in the program were evaluated against the ability of an insider to successfully commit radiological sabotage resulting in a danger to the public's health and safety.

NEI Comment: The use of danger as a criterion is undefined and open to interpretation. The NRC is in the process of defining the term "radiological sabotage." The standard should be risk-significance. A clear definition of risk-significance and radiological sabotage must be part of any definition needed for this inspection area. The evaluation should be the result of using the agreed upon SDP.

(4) root cause analyses (if required depending on risk-significance) and corrective actions were timely and adequate to prevent recurrence.

NEI Comment: Is there a clear standard for significance related to these problems? If not, a clear definition of risk-significance is needed for this inspection area.

(6) required reports were made to the Commission or inputs to the correct PI were made.

(7) the performance trend indicated by the sample set was consistent with the performance levels reported by the licensee for the applicable PI.

NEI Comment: It is not clear what is expected here. The expectations should be clarified.

During the inspection, note whether issues that occur are evaluated for identification to a corrective action system and that significant issues are input to the system.

NEI Comment: Is there a clear standard for significance related to these problems? If not, a clear definition of risk-significance is needed for this inspection area.

Verify that corrective actions have been properly implemented for five risk-significant issues identified since the last inspection.

NEI Comment: Is there a clear standard for significance related to these problems? If not, a clear definition of risk-significance is needed for this inspection area.

<u>Response to Contingency Events (Protective Strategy and Implementation of Protective Strategy)</u>

INSPECTION BASES: This is a risk-significant system necessary to protect against the external Design Basis Threat (DBT). The licensee should be able to demonstrate the ability to respond with sufficient force, properly armed, appropriately trained, and within the appropriate time to protected positions to interdict and defeat the design basis adversary force in order to **protect vital equipment necessary for the safe shutdown of the plant. The consequence to radiological sabotage if an attack is successful is high.**

NEI Comment: Replace with the highlighted section with the following:

"interdict the design basis adversary force in order to prevent a Part 100 Release. The consequence to radiological sabotage if an attack is successful is low due to engineering controls and Operations' capability to mitigate damage." There is no requirement to defeat the adversary force.

LEVEL OF EFFORT: This inspection effort will be accomplished every two years using approximately 104 hours of direct inspection effort onsite. As determined by performance, events, and regional staff, this baseline inspection effort may be completed anytime during the inspection period. This inspection effort will be managed by regional staff and generally led by regional inspectors. The composition of multi-person inspections **may include regional security inspectors (including from other regions), headquarters specialists and contractors capable of evaluating response, alarm and assessment equipment, small unit tactical response and defensive strategies, depending on the scope of the inspection. However, every six years each plant shall be inspected using a team composed of regional security inspectors, headquarters specialists, and contractors.** This inspection will also satisfy the annual requirement for the Performance Indicator for the Physical Protection System.

NEI Comment: This inspection is supposed to be performance-based and not tied to specific periodicity or hours of inspection effort. "Tactical" response is not used in regulation for NPP. The inspection manual does not clarify the inspection differences between these two types of inspections.

INSPECTION OBJECTIVE: Verify through licensee drill and NRC-evaluated exercises that the licensee's protective strategy works, and it can protect its vital area target sets against the design basis threat.

NEI Comment: Need to revise by adding the words "records, observations, and self-assessment" between "drill" and "and." Substitute "meets requirements" for "works." Delete "its vital area target sets."

INSPECTION REQUIREMENTS:

B. Conducting the Inspection:

NOTE: When conducting the inspection, the inspectors should take the approach that an adversary must first penetrate the protected area intrusion detection system by a covert or overt action. If the inspectors determine that it is possible to enter the protected area through a covert action then the inspectors should consider the defensive positions that the adversaries would have to pass to reach a target. If the adversaries can reach a specific target set without alerting the security force, the licensee has a significant vulnerability that requires immediate compensatory actions.

NEI Comment: Revise by inserting the words "(as defined in the physical security plan (i. e., jumping, running, crawling, etc.)" between "action" and "then." Also insert the words "and CCTV camera" between "that" and "the."

1. Intrusion Detection System:

   The second inspector should accompany a licensee supervisor and individuals who normally conduct testing of the perimeter intrusion detection aids on a tour of the perimeter to establish potential vulnerabilities that could be penetrated by individual(s) undetected and used as potential routes of travel to target sets. Only those areas specifically identified will be performance-tested by the licensee at the request of the inspector. The inspector should select no less than three locations for crawl testing, three for simulated jump testing, and three for walk testing. The licensee should be able to demonstrate that if an area seems susceptible to jumping that the zone in question cannot be jumped. The licensee can accomplish this by demonstrating a simulated jump using a device such as an aluminum ball, **12 inches in diameter, being passed over the zone at a height of six to seven feet.**

NEI Comment: The specification of the 12- inch ball for the jump test is not consistent with past NRC practice at plants. Further, there is no requirement in this area. This inspection guidance is a further example of regulation through inspection in the Security area. Revise the highlighted section as follows: ". . . 12 or 26 inches in diameter, being passed over the zone at a height sufficient to prevent jumping across the zone of detection without generating an alarm. This distance will be different based on how the zones are configured and if other measures have been employed to prevent jumping." It is not clear if this includes Perifields in that the top wires are about 12 feet high and not vulnerable to being jumped over.

The inspector in the alarm station should be able to recognize and identify the individuals in the detection zones on the assessment monitors. If the assessment aids or monitors are not clear, verify the licensee has compensatory measure in place for degraded equipment.

NEI Comment: Add the following sentence at the end of this paragraph: "Environmental conditions such as rain or fog will be considered when testing and compensatory measures specified in procedures will suffice for successful assessment." It is not clear what predictable or exploitable criteria the inspector uses to determine the need for compensatory measures if the "monitors are not clear."

Require the licensee to demonstrate its capabilities to respond and to effectively implement the defensive strategy and response procedures. The number, type, complexity, and focus of the demonstrations will be determined based on site-specific considerations, such as topographical layouts, nature of target sets, and previously identified performance. **The NRC inspectors will choose target sets to be defended, provide a profile of the adversary (within context of the DBT), provide a description of adversary equipment to be simulated, and specify points of adversary entry to the protected area.** The licensee's response should include only those capabilities outlined in the licensee's security plan, protective strategy, and implementing procedures.

NEI Comment: Revise the highlighted section as follows: "The NRC inspector will review the licensee's self-assessment program and contingency response strategies. The inspector will observe various aspects of the program and licensee-evaluated exercises to validate program adequacy." This change is appropriate to allow the industry initiative to substitute for the current force-

on-force evaluations and lay the groundwork for a performance indicator which will be developed in the future.

A licensee's response to a scenario should be considered adequate when **they interdict the adversaries: (1) in a timely manner; and (2) with sufficient numbers of responders who are appropriately armed and in protected positions**.

NEI Comment: Replace the highlighted section with the following: ". . . they prevent radiological sabotage resulting in a Part 100 Release by engaging the adversaries: (1) in a timely manner; (2) with sufficient numbers of responders who are appropriately armed and in protected positions; and (3) including mitigation of damage to plant equipment by Operations Personnel.'"

Should a licensee demonstrate that they are **not able to provide the minimum response required**, it should be requested to take compensatory measures which will ensure that the minimum response needed to provide protection against the DBT is effectively implemented. Inspectors will verify the implementation of that compensatory action before they leave the site.

NEI Comment: Replace the highlighted section with the following: ". . . not able to prevent radiological sabotage resulting in a Part 100 Release, ..."

C. <u>IDENTIFICATION AND RESOLUTION OF PROBLEMS:</u>

(2) the **risk-significance** of the findings was properly addressed.

NEI Comment: Is there a clear standard for significance related to these problems? If not, a clear definition of risk-significance is needed for this inspection area.

(3) root cause analyses (if required depending on **risk-significance**) and corrective actions were timely and adequate to prevent recurrence.

NEI Comment: Is there a clear standard for significance related to these problems? If not, a clear definition of risk-significance is needed for this inspection area.

(6) the **performance trend indicated by the sample set was consistent with the performance levels reported by the licensee for the applicable PIs**.

NEI Comment: It is not clear what is expected here. The expectations should be clarified.

Response to Contingency Events Inspection Guidance Table Examples:

> Intrusion detection systems have no obvious vulnerabilities.
>
> Security responds with sufficient force, properly armed, appropriately trained, and within appropriate timeframe to interdict and defeat the design basis threat.

NEI Comment: Replace the two examples as follows:

> *Intrusion detection systems have no obvious exploitable vulnerabilities.*
>
> *Security responds with sufficient force, properly armed, appropriately trained, and within appropriate timeframe to prevent radiological sabotage resulting in a Part 100 Release.*

<u>Management Directive MD 8.3</u>

Some of the criteria for initiation of an IIT seem to be low and subjective:

- Operation outside the design basis (especially since NEI and NRC have not agreed on the cutoff level of detail for what constitutes design basis information).
- Possible generic implications (needs a hard threshold level tied to SDP).
- Repetitive failures or events (needs a hard threshold level tied to SDP).
- Questions/concerns pertaining to licensed operator performance (need guidance on how to extrapolate performance assessments not related to events i.e., training or exam deficiencies, to an SDP evaluation. Similarly need guidance on how to extrapolate single crew performance, e.g., exam, training or event, to entire staff).
- Footnote ** should establish the color level at which an IIT would be initiated. The current wording is ambiguous and seems to allow an IIT that uses the white supplemental response procedure. Need better definition of rules of engagement to avoid surprises or confusion.

Comments on Column 4 Procedure

Page 2, 2nd bullet indicates this procedure is not intended for event follow-up. NRC needs to clarify how this procedure relates to the color assigned to an event that is processed through the SDP. It is useful to note that the specific baseline procedures are used for initial event follow-up; however, this procedure may be used to assess the implications of events that are very risk significant (single red finding).

The inspection objective described in 01. 01 needs clarification. The wording implies that a column 4 inspection is designed to see if the plant should move into column 5. NEI does not believe that the number one objective is to do that. NRC needs to be clear about the purpose of the procedure. Is the purpose of the procedure fact finding to support management decisions required by the action matrix that are made elsewhere? If so, then NEI would suggest that 01. 01 be reworded to say that more directly. If, on the other hand, the procedure is intended to provide the framework for the action matrix decision, then it must be expanded to link specific conclusions that might be drawn to the appropriate action matrix regulatory action (i.e., DFI, 50. 54(f), CAL, or Order).

The action matrix and the procedure do not describe the actions that might follow from a significant event (red) that is judged to be a new industry issue rather than a plant performance problem. It needs to cover this possibility if the procedure is intended to be used to evaluate event-related single red findings.

If one of the possible regulatory actions that can be taken in column 4 is to move a plant into column 5, then NRC needs to be straight forward on this decision. The action statement should be revised to indicate this potential. The column 4 procedure and or decision making process will also need to have the established criteria for moving an item into column 5. Overall, this aspect continues with the overall perception that the NRC view, as manifested in the column procedures, is to be shifting everything one column to the right.

On the positive note, the key attributes cover the kinds of things you would want to look at to find the underlying drivers to the problems that resulted in the colors.

The discussion in 02.03A.3 does not flow with what follows. It looks like 02.03A.4 got jammed in the middle. Also, the connection between the number of the attributes elements in 02.03 and the correlation with similar numbering in 03.03 is not obvious. Similarly, the use of the same topic headings for 02.XX and 03.XX is confusing when trying to understand what the section is trying to convey.

NEI hopes that all the hours listed on page 29 (~9 man-months) are not charged to every use of this procedure.

Section 3.02.B, (page 9). This section has two problems. The first is the first sentence which looks at licensees capability to "identify performance issues before they result in actual events..." This a very subjective call and is easy to conclude in hindsight that you should have known. This needs to be replaced with a more objective criteria or reword these sentence to say that "the critical elements of a program to identify....are in place". Same section, last sentence implies that management's support of audit and assessment is linked to QA organization staffing. Staffing of QA is not an indicator and should not be listed here.

Section 3.0.2.E (page 10), regarding the employees concerns program, asks the question "...additional safety issues exist that have not been adequately captured by the corrective action program..." Again, NEI believes that utilities may resolve those issues under the Employee Concerns program without entering them into the Corrective Action Program (for confidentiality reasons, etc.). The statement should be something to the effect that the concerns were adequately addressed and resolved.

Comments on Manual Chapter 0350

Need to clarify the entry conditions defined in the first paragraph on page 2. Voluntary shutdowns by a utility ahead of significant performance degradation (as measured on the action matrix) should not result in the use of Manual Chapter 0350. NRC needs to avoid the punitive situation where the voluntary shutdown contributes to the indicators degrading further. This situation could happen if a utility proactively shutdown after an event (i.e., trip, FET, etc.) that that occurs early in the quarter. The shutdown to investigate and resolve the problem can cause the PI to further degrade if the quarter that rolls of the PI had a lot of critical hours (but no event data points) and the current quarter utility has fewer critical hours. We do not want to set up a situation where a utility must choose to operate and accumulate critical hours to avoid invoking Manual Chapter 0350.

Need to clarify the entry condition discussion in the second paragraph on page 2 and the first paragraph of section 05.02. The procedure blurs the distinction between column 4 and column 5 and applied this procedure to either case that involves and extended shutdown. It would be clearer to define the entry conditions for column 5 on the action matrix and apply

Manual Chapter 0350 to column 5. This would eliminate the confusion of whether the column 4 supplemental inspection procedure applies to column 4 or Manual Chapter 0350.

Similarly, the first paragraph on page 5 outs the focus of the restart review on the column 4 entry conditions. The same clarification is needed here.

The wording of bullet 2 on page 5 can be interpreted to require the utility to completely resolve design and licensing basis issues even though they may be evaluated as green by the SDP. Is that what is intended? Note that the paragraph after the bullets sets the threshold for restart issues as any non-green finding.

It also seems inconsistent to set the threshold for restart issues as any non-green finding. White findings are considered of low significance and are only assigned follow-up inspections for operating plants. The threshold for full resolution for restart should be set at yellow and red findings.

Comments on Operating Reactor Assessment Program

Page 5 of 25 - Regarding discussion of Problem Identification and Resolution in the annual agency review and assessment letter, it is imperative that NRC and the industry have agreement on the purpose of the PI&R inspection module, the method of collating results and drawing conclusion, and format for presenting the results. NEI suggests that it would be imprudent to experiment with this subject in the important public forums.

Pages 7/8 of 25 - The criteria for a column five classification seem reasonable.

ENCLOSURE 4

Comments on Significance Determination Process (SDP)

Overall Impressions

In general, the SDP processes for Reactor, EP, and RP appear to be well thought out and suitable for implementation, with some changes as discussed below. The pilot process has demonstrated that a solid working relationship between the utility PRA staff and regional SRA staff is necessary to ensure that risk can be appropriately assessed and decisions made in a timely manner to support plant assessment. This professional working relationship is beneficial to both and is encouraged for full program implementation.

The SDP process is still evolving for the Event SDP and the Shutdown SDP. Additional analysis, pilot testing and review will be necessary before implementation.

The Fire protection and Physical Protection SDPs are not yet ready for full implementation. The Fire Protection SDP has not yet been fully tested and therefore has not been proven effective in assessing fire risk. The Fire Protection SDP has not been fully tested. The Fire Protection SDP reviews at the two pilot plants were performed using a draft inspection guidance document that was not available to the pilot plants during the inspection and still has not been formally issued (See Section 5.0 Step 2&3 of the SDP). This draft inspection guidance was provided to the NRC team who was using the guidance to evaluate the inspection findings. This document provides the guidance on how to characterize the degradation of the defense-in-depth (DID) features of fire protection. Determining degradation of DID features is also subjective - requires discussion between the NRC and licensee to agree on the appropriate level of degradation. The determination of fire protection feature degradation is very subjective and is dependent upon the individual inspectors opinions. Numerous questions will need to be answered before the industry understands how it is intended to work. It contains significant new aspects that will need to be discussed with the industry in public meetings in order for industry to understand the process and determine whether it includes requirements that exceed the current regulations.

The security SDP lacks sufficient guidance to result in repeatable results, and overemphasizes situations in which there is no significant increase in the likelihood of damage to the reactor, making it inconsistent with the other PI thresholds and SDP findings. A proposed change to the Security SDP was developed during a public meeting held December 21, 1999 that, if accepted, addresses this concern.

Additional Remaining Issues

The industry and NRC had very limited opportunities during the pilot process to deal with non-green findings (white, yellow, or red). This should be factored into the planning for full implementation. Some specific issues need further work:

- The process to evaluate and characterize non-green findings is not well defined and understood. The result of this was that the NRC's basis for preliminary results were not made available to the affected utilities in a timely manner. Similarly, the process for utility interaction was not clear which led to delays in the affected utility's being able to engage the evaluation and provide their perspective on the issue. While the clarification provided in the recent guidance memorandum for the pilot program (EGM - 006) appears to address these issues, this still needs to be factored into the plan for full implementation.

- Some problems were also encountered in implementation of non-green findings. Specifically, assumptions used in the NRC Phase Two and Phase Three analyses were not documented in a timely manner. These delays hampered the utility's understanding of the NRC's perspective on the issues and made utility/NRC interactions less efficient and effective.

Terminology is not consistent across the cornerstone SDPs. It appears that the terms issue and finding mean different things in different SDPs. This leads to difficulties in communications and reaching common understanding.

Development of plant-specific reactor safety SDP work sheets for non-pilot plants needs to be completed.

NRC should formalize a change process for major changes to the existing SDPs and any new SDPs (i.e. shutdown SDP). The industry believes that it is imperative that the process includes opportunities for stakeholder review and input as well as a pilot process to ensure any ensuing problems can be addressed prior to finalization.

The NRC "Pre-Screening" process as provided in Manual Chapter 0610* appears to be adequate with respect to the Reactor Safety Cornerstone. Similar processes need to be developed and validated for the remaining cornerstones.

The treatment of external events within the SDP process needs further attention. The provision for increasing the event frequency as outlined in the guidance is probably inappropriate for external events. It should be recognized that such application can result in non-green findings even though the plant is in full compliance with its licensing basis. This raises concerns regarding changing licensing requirements by inspection without application of the backfit rule requirements. Industry believes that it would be more appropriate to focus on the mitigation capability impacts using the event frequency fractions established in the IPE/IPEEE work.

Manual Chapter 06XX Revision 1, dated 10 August, 1999

<u>General Comments</u>

The definition of the word "resolve" should be placed within the body of 06XX to ensure it is being consistently applied across all Cornerstones.

The NRC should look hard at the treatment of external events for lessons learned. Namely the practice of increasing the event frequency as outlined in the SDP guidance is probably not appropriate for external events since it is outside the control of the utility. Instead, the SDP evaluation should focus on the mitigation capability impact using the event frequency fractions established in the IPE/IPEEE work.

NRC should recognize that the treatment of external events can lead to non-green findings even though the plant is in full compliance with its licensing basis. This raises an interesting dilemma for a white finding in this area, which represents a minimal reduction in safety and therefore is below the level that would pass the backfit test.

<u>Reactor Safety SDP</u>

In general this SDP appears to be well developed and usable. Experience during the pilot process has resulted in generally consistent application and results. An important action to be completed will be the development of the plant specific SDP for each non-pilot plant prior to full implementation.

Draft plant-specific PRA models for the phase 2 screens were developed for each of the pilot plants by the NRC. These models were commented on by the respective plant PRA groups to identify differences between the draft and the current plant model. There can be significant differences between the two. The NRC and the industry need to agree on how to provide an up to date PRA model to the NRC to ensure an accurate reflection of plant risk is maintained.

For those few issues that have undergone a Phase 3 assessment, it appears that this process is not timely. This is a concern for full implementation where it can be anticipated that more of these reviews will be required. Problems were experienced with the process used to evaluate a potential non-green finding. The process to interact with NRC on the finding during phase 2 and phase 3 were not well understood. The basis for the NRC preliminary results were not made available in a timely manner. Similarly, the process for utilities to provide its perspective on the finding were not well understood for too long. The guidance in the recent enforcement guidance memorandum for the pilot program appears to address the process issues.

Additional problems were encountered with the process used to evaluate a potential non-green finding. The assumptions used in the NRC phase 2 and phase 3 analyses were not documented in a timely manner. The delay hampered the utility's understanding of NRC's perspective on the issue. It also made the interactions with NRC less efficient. NRC should ensure that key assumptions it makes in the preliminary evaluation are properly documented in a timely manner and shared with the utility.

A formal guidance document needs to be prepared to identify the process for requiring a Phase 3 screen and the rules under which the utility and the NRC will operate.

<u>Comments on specific pages:</u>

Page A1-1, Defining Characteristic, the description of the "most important intended characteristic" needs to be modified to state:

> "The most important intended characteristic of this process is that it provides a consistent objective assessment of risk significance of issues for inspectors and their management."

Page A1-3, Process Discussion, in the second paragraph, the term "controversial findings" is not defined. As a result, it appears that this term could apply to findings of any color, including those screened as green, and could result in green findings being required to undergo a Phase 3 evaluation.

Page A1-5, in the first paragraph, modify the last "OR" section to read as follows:

> "...OR the finding has not resulted in the loss of function of a non-Tech-Spec controlled risk-significant system, structure or component under the maintenance rule (10 CFR 50.65) for greater than 72 hours."

The substitution of "loss of function" for "failure" is consistent with the usage for Tech-Spec systems. The substitution of "72 hours" for "24 hours" is more consistent with Technical Specifications Allowed Outage Times *and the Performance Indicator definition for Unplanned Power Changes >20%.*

Page A1-6, Step 2.2, paragraph 3, There needs to be a clarification and guidance on values and bases for increasing the frequency of events listed in Table 1.

Page A1-9, Step 2.3, the use of standby liquid control in BWRs should not be considered an operator action under high stress. This is a very simple action requiring the use of one switch at most BWRs and should be considered an Operator Action for the purposes of this SDP.

Page A1-9, Step 2.4, the guidance is not clear as to how the "remaining mitigation capability" ratings are summed for each scenario and used with Table 2. Provide more guidance within Step 2.4 to show how this process is to be performed.

Page A1-10, Table 2, Recommend re-naming the "Remaining Mitigation Capability" as "Scenario Total Remaining Mitigation Capability".

Emergency Preparedness SDP

1. EP SDP – General – The SDP colors (Green, White, Yellow, and Red) are equated to the violation severity levels (Level IV, III, II, and I). The implementation of the EP SDP process would result in more violations than with current NRC processes. Need to evaluate the new risk based inspection process proposed to ensure that it is not more restrictive.

   The SDP is an excellent screening tool. Some amount of developmental work is needed for a risk-based process to more closely examine the risk significance of each issue. Using the current flow chart, a "yellow" EP issue may not be equal in significance to a "yellow" issue in other cornerstones.

   Consistent with the objective of the EP cornerstone, each issue should be reviewed for whether it affects the licensee's ability to take adequate measures to protect the public health and safety.

   The internal plant events SDP is used as a "screening tool" to determine whether to screen out issues of low significance (green issues). Likewise, the EP SDP can be used as a screening tool in a similar manner. If an issue turns out to be other than "green," then a more detailed evaluation can be performed for the specific issue or circumstances.

   For example, for a general emergency during an actual event, it is easy to recognize that there is a difference in significance between a failure to notify by one minute (i.e., missed 15-minute goal by one minute), and a complete failure to notify. Whereas the first issue has essentially no significant impact on the measures taken to protect the public health and safety, the latter clearly could. Duration of the failure to notify is a key factor in characterizing the significance of the issue.

   Perhaps duration time criteria could be established to provide some measure of risk significance. The issue would first have to be shown to have an actual or likely impact on the measures taken to protect the public health and safety. The impact could then be measured or estimated and placed in a time interval. For example, a conceptual model might characterize a relatively short delay in notification as green, and a complete failure to notify or significant delay in notifying as red.

2. EP SDP Sheet 1 - A screening process for inspectors needs to be formalized for the EP SDP process to keep every issue from ending up in the SD process.

   Are the entry conditions clearly defined in any guidance? What is meant by a "violation?" Does it include all cited, non-cited, and minor violations?

   Is "Finding Identified" the best wording for entry into the SDP.

3. EP SDP Sheet 2 – The criteria for resetting the clock for failures to meet RSPS and PS needs to be formalized. What is meant by a failure to meet a PS or RSPS? Are there any criteria or can examples be given?

   With regard to failures to meet multiple standards, need to clarify that failure to meet an RSPS would not also count as failure to meet a PS. May want to state in 06XX that the standards that are not RSPS compose the PS.

4. EP SDP Sheet 3 - Is the supporting guidance clear that failure to implement means failure to notify? NRC discussions noted that missing the 15-minute goal of classifying or notifying would not meet the intent of failure to implement. What is the threshold?

5. EP SDP Sheet 4 – Is the word "resolve" being used consistently in all SDP's with regard to corrective action programs?

   The criteria for failure to resolve or to resolve in a timely manner seems to focus on ease in which the corrective action can be implemented. For example, a criterion of 60 days is given for a corrective action taken in response to a failure to meet an EP related requirement, whereas 14 days is the criterion for failure to implement as EP related requirement.

   Corrective action program guidance for timeliness typically focuses on the significance of the issue, not simply the complexity of the corrective action. For example, corrective actions being taken to correct or prevent a significant condition adverse to quality should generally be more timely than corrective actions taken to correct a minor condition adverse to quality.

   Timeliness concerns should be tied to significance of the issue. In the absence of clearer criteria for timeliness, one possible means for handling such concerns would be for EP inspectors to "flag" such concerns for further review during the corrective action inspection module.

6. <u>EP SDP Sheet 1 and 4</u> – After reviewing all other SDP's it appears that the EP SDP emphasis on the timely resolution of items placed PIDR/CAP is not consistent. A review needs to be performed to ensure that this criterion is not overly restrictive and is consistent with other processes.

7. <u>EP SDP Sheet 5</u> - If a licensee fails to critique a misclassification of an NOUE on two drills, sheet 5 would classify the drill/exercise problem as "yellow." Such a classification is not on the same level of significance as a "yellow" on other PIs.

   One way to consider repeat failures is in the NRC's review of the corrective action program, which should be designed to prevent recurrence of a significant condition adverse to quality. Corrective action programs should be designed to separate significant conditions adverse to quality from conditions adverse to quality. The level of causal analysis should be commensurate with safety significance. For example, a significant condition adverse to quality would receive a root cause analysis, whereas a condition adverse to quality would typically receive only an apparent cause analysis. Corrective actions for conditions adverse to quality are typically intended to fix the immediate problem but are not typically designed to prevent recurrence. Thus, repeat problems alone are not an indication of a "broken" corrective action program. Issues of repeat occurrence should be "flagged" for review during the corrective action program inspection module.

8. <u>EP SDP Sheet 1, 4, and 5</u> –Need to determine if "Inspection/Exercise Observation" is the best wording for conditions that do not that go through the SDP and do not even warrant being GREEN. In addition, need to provide guidance as to how this information will be consistently conveyed to the utility.

9. <u>EP SDP Sheet 4 and 5</u> – If a utility self-identifies an issue that is a failure to meet a PS and this issue is placed in the PIDR/CAP for resolution, will this issue be evaluated via the SDP by the inspectors and can it result in a green (or worse) finding?

<u>Occupation Radiation Safety SDP</u>

ALARA Findings

(Page A2-9) The discussion on "ALARA Findings" needs to be updated to reflect the revised SDP. The text is currently based on the 8/10/99 version of the SDP, rather than the most current 11/12/99 version that is shown.

(Page A2-13) Separate "actual job dose" criteria are shown in the fourth and fifth blocks, i.e., for PWRs and BWRs. The job-dose values have been derived as 4% and

20% values of the baseline collective dose values that serve as screening criteria (third block). We do not believe that the approach of using separate criteria for PWRs and BWRs is valid for the job-dose values because they implicitly represent criteria for a determination of relative dose-significance, rather than serving as a performance benchmark. The SDP should employ a single job-dose value in each of the blocks. We suggest that the two values in each block be averaged and rounded to a single digit, yielding 10 rem and 40 rem, respectively.

(Page A2-13) An applicable time period should be specified for the block, "Greater than 2 occurrences?" We suggest that an appropriate time period is "in the assessment period" (e.g., per year). This would be consistent with the approach taken in the public radiation safety SDP for radioactive material control.

Exposure Control Findings

General comment: The process for initial screening of items prior to entering the SDP in the area of occupational radiation safety is not well defined and understood. Explicit screening criteria should be provided similar to the screening criteria that are included in the ALARA SDP. Items of negligible safety significance and little or no potential for any consequence (i.e., with regard to radiation dose to workers) should be screened out as "observations," and not be entered into the SDP process with the result of becoming green findings. We suggest that such criteria screen out items that do not involve any of the following:

- Unintended exposure
- Substantial potential for overexposure
- Compromise of the ability to asses dose
- Violation of a regulatory requirement (e.g., 10 CFR Part s 19 or 20)

(Page A2-10) The SDP should include guidance to clarify that if an "unintended exposure" occurrence has been documented as a PI event, and also does not constitute an overexposure or a substantial potential for overexposure, it will not be documented as a green finding. If already documented as a PI event, the item will already have been placed into the licensee's corrective program, and "double-counting" as a green finding will be non-productive and potentially misleading.

(Page A2-10) The discussion of "unintended exposure" should be revised to improve clarity and consistency with the performance indicator (PI) for occupational radiation exposure control. The first paragraph characterizes any unintended dose that exceeds the exposure that exceeds the criteria in PI as "significant," which is potentially misleading and inconsistent with the SDP.

First, the discussion of the PI criteria (NEI 99-02 –Draft Revision D) is clear that: "the dose criteria are established at levels deemed to be readily identifiable, based on industry experience. The dose criteria should not be taken to represent levels of dose that are 'risk-significant'. In fact the criteria are generally at or below dose levels that are required by regulation to be monitored or to be routinely reported to the NRC as occupational dose records."

Second, the SDP would screen unintended dose occurrences at the levels of the PI criteria as "green," which is by definition not significant.

(Page A2-12) The SDP chart should be revised to improve its internal consistency. The blocks for actual overexposures (i.e., consequences) have been appropriately derived from previous enforcement criteria at 1x and 5x the regulatory limits. In contrast, the blocks for events that involve a potential for overexposure lead to illogical conclusions regarding significance.

For example, an unintended dose occurrence that does not exceed the regulatory limit would be "green," based on consequence. However, if the event occurred in an area with dose rate levels >25 R/hr, the event would be ranked as "yellow," which is comparable to an overexposure. Further, the criterion of >25 R/hr lacks a firm basis in either historical performance or in implied significance. Also, the potential "red" finding associated with a "substantial potential" occurrence in a very high radiation area that does not involve an actual overexposure is not consistent with either the consequence-based blocks in the SDP or the bases for criteria in the enforcement policy.

We recommend that the "Area >25 R/hr" block be deleted, and that the finding associated with a "substantial potential" occurrence is "white" if it is not associated with a very high radiation area, and "yellow," if it is.

Public Radiation Safety SDP

Public Radiation Safety (Rad Material Control, Effluent Release Program, and Environmental Monitoring Program)

(Page A2-17) Clarification should be provided that the dose values given in the SDP refer to the total effective dose equivalent (TEDE).

(Page A2-17) The SDP should include guidance to clarify that the dose-based criteria for public exposure explicitly do not apply to discrete radioactive particles. The presently available methods for estimating exposures from discrete radioactive particles do not reflect the current scientific understanding of potential health risk from such exposures. Discrete radioactive particles do not pose any substantive risk at the dose levels included in the criteria in the SDP because any resultant dose is highly localized. This has been concluded in extensive research conducted by NRC and others, as well as in reports of the National Council of Radiation Protection and Measurements (NCRP). The scientific understanding of the negligible health risk posed by discrete radioactive particles has also served as the basis for Commission approval of proposed rulemaking to revise regulatory requirements for estimating controlling exposures from particles.

Transportation and Part 61

Guidance should be provided to clarify the regulatory bases and applicability for the SDP on Transportation and Part 61 (there currently is no guidance). For example, there is no apparent regulatory basis for the significance determination criteria in the section on "Low-Level Burial Ground (LLBG) Access Problem, nor is it clear what is meant by a "problem" that is not associated with "denial of LLBG access" or "Part 61 waste underclassification."

In the Certificate of Compliance (COC) section of the SDP, the meaning of the decision blocks on "minor contents deficiency" and >1 critical contents deficiency" should be clarified.

In the section on "Radiation Limit Exceeded, "the logic flow should be revised to reflect the possibility that both the external radiation levels and the surface contamination levels criteria could be exceeded.

Physical Protection SDP

This SDP should be replaced with the version developed during the December 21, 1999 public meeting.

Fire Protection SDP

General

This SDP is more complex and less user friendly. It does not appear that the screening of deficient conditions would produce results that are consistent with the results that would be expected from the Reactor Safety SDP.

The credit for fire brigade actions and /or effectiveness does not appear to be consistent. The positive contributions of fire brigade intervention are discounted while fire brigade performance deficiencies can affect multiple schemes.

Clearly, the SDP developed for fire inspections is not risk informed but deterministically based. This philosophy is not consistent with other SDP modules developed to date and is not even aligned with the Commission direction to risk-inform the regulatory process.

The SDP has an FMF equation with the first factor being the Ignition Frequency of a Fire. This number is a probability number with an order of magnitude (Ex: 3.9E-3) that is converted to an integer value. However, the SDP does not describe how to convert this number. During the Salem inspection exit, the NRC Inspectors explained the use of the logarithm conversion method. This conversion method needs to be explained in the SDP. Additionally, this discussion should have occurred before the inspection commenced.

As discussed with the Region I SRA, the fire ignition frequencies quoted in the SDP are very conservative and may need to be adjusted to make the Phase II review a useful process.

Comments by Section

Section 1.0, Introduction

Footnote 1 states "Fire protection features sufficient to protect against fire hazards in the area, zone, or room under consideration must be capable of assuring that necessary structures, systems, and components needed for achieving and maintaining safe shutdown are free of fire damage (see section III.G.2a, b, c of Appendix R to 10CFR Part 50); that is, the structure, system, or component under consideration is capable of performing its intended function during and after the postulated fire, as needed." This is a mis-representation of the regulatory requirements. Appendix R III.G.2 requirements are only invoked for those plant areas that cannot show they already satisfy Appendix R Section III.G.1. This is a subtle but important distinction, since barriers, detection, suppression, etc. credited under III.G.1 would have received prior NRC review and approval under BTP APCSB 9.5-1, or BPT APCSB 9.5-1, Appendix A, and do not need to meet III.G.2 criteria (ex., 3-hour barriers may not be required).

Footnote 2 states "An SSD success path **must be capable of maintaining the reactor coolant process variables within those predicted for a loss of AC power**, and the fission product boundary integrity must not be affected (i.e., there must be no fuel cladding damage, rupture of any primary coolant boundary, or rupture of the containment boundary)." The first portion of this statement is a backfit. These criteria are taken from 10CFR50 Appendix R, section III.L, which is only applied to Alternative Shutdown capability. Application of these criteria in the SDP effectively takes redundant safe shutdown paths away from the licensee when the SDP is utilized, resulting in an overstatement of the significance of a finding.

Section 4.0, Fire Protection Risk Significance Screening Methodology - Phase 1

This section refers to plants' Fire IPEEEs as PRAs, however in many cases, a screening methodology such as EPRI FIVE have been utilized. When a screening approach is utilized, results for individual compartments have a tendency to "clump" around whatever screening value was selected, and are not truly representative of a particular magnitude of fire risk, since they may have screened at different phases during the analysis. Fire IPEEE results are also not indicative of a "Final CDF" for fire events, since they typically only look at a limited subset of all plant mitigating equipment, and do not consider all possible recovery scenarios included in IPEs. . In light of this, the fire IPEEE results should be considered conservative representations of CDF.

Step 1: Screening of Fire Protection Findings
"Making judgements regarding how effective a fire brigade can be in extinguishing a challenging plant fire requires an evaluator to have a comprehensive understanding of manual fire fighting techniques and operations." This criteria places too much emphasis on inspector's judgement. Criteria for Fire Brigades are available in the codes of record for the facility, the UFSAR, commitments, and SERs. In most cases, these sources should be definitive enough such that "expert judgement" need not be the sole determinator of fire brigade effectiveness. For "Risk Based" applications such as the SDP, a measure of fire brigade effectiveness would also be whether the fire brigade response is consistent with that modeled in the IPEEE for the facility.

Footnote 3 states "Allowed outage times with the use of compensatory measures do not provide an equivalent level of fire safety to that of a fully operable fire protection system or feature. Long-term use (more than 30 days) of compensatory measures for degraded or inoperable fire protection features used to protect safe shutdown capability is an indication of inappropriate attention and resources being given to managing fire risk vulnerabilities." There is not a clear basis for statements made regarding the use of compensatory measures being less effective than installed equipment or barriers. Nor is there any basis for the greater than 30 day criteria. **This item is a backfit**. The NRC has found on several previous occasions that degradation of one element of the defense-in-depth fire protection posture of a plant does not typically result in an inordinate increase in plant fire risk[1] [2]

[1] "Vermont Yankee Power Station; Issuance of Final Director's Decision Under 10 CFR 2.206" [Federal Register: January 7, 1998 (Volume 63, Number 4)] - "LER 96-18: Inadequate installation and inspection of fire protection wrap results in plant operation outside of its design basis; a single fire would impact multiple trains of safety-related equipment".

[2] "All Licensees of Reactors With Installed Thermo-Lag Fire Barrier Material; Issuance of Director's Decision Under 10 CFR 2.206" [Dated April 3, 1996] - "Every licensee with Thermo-lag fire barriers will continue to maintain NRC-approved compensatory measures, such as fire watches, until permanent corrective actions are implemented. Therefore the public health and safety are protected." "Generally, therefore, by providing additional fire prevention activities through enhanced detection capabilities to find fire hazards and in the case of a fire, augmented suppression activities before a barrier's ability to endure a fire is challenged, fire

In various SERs NRC has reviewed and already approved the conditions of Appendix R Safe Shutdown Program, which include statements that effectively equate that a firewatch is equivalent capability for the degraded equipment. However, in this screening process, the NRC is concluding that they are not equivalent. This is an inappropriate way to change requirements for compensatory measures by raising the standards on which they are measured. This is the basis for the original reliance on fire watches as a compensatory measure for those fire protection features governed by Technical Specifications. Although licensees have removed fire protection from technical specifications, the requirements for fire watches remain unchanged.

This has been further evaluated at some plants as part of their IPEEE. For example, at several plants, the base IPEEE analysis took no credit for raceway fire proofing materials, and still were able to show an acceptable fire risk profile. In addition, the NRC has provided testimony in the Thermolag court case stating that fire watches were an adequate replacement for installed equipment and or barriers to protect the health and safety of the public (Refer to DD-93-3). In general, specifying a 30-day time limit to conceptualize, design, and install corrective actions is not realistic for many plant changes, in fact placing undue urgency on such changes may result in the performance of corrective actions that are detrimental to overall facility safety. Since both the fire protection and safe shutdown SSC's may interact with safety-related equipment, changes to these systems should be carefully considered.[3]

<u>Step 2: Safety Importance Determination</u>
Figure 4-2 states "SSD system with redundancy (e.g., **all high pressure reactor inventory control functions**) is location in the area, zone, or room of concern." This figure perpetuate the error that only high pressure systems are "eligible" for use for post-fire safe shutdown. Continuation of this error throughout the SDP results in many cases where no credit is given for safe shutdown "paths" that are in fact available for use. This ultimately results in a significant overstatement of significance under the SDP. This ultimately results in a significant overstatement of significance under the SDP. This concept is not consistent with the "at power" SDP which considers redundancy and diversity of all systems capable of providing injection.

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watches compensate for degraded fire barriers." "In sum, notwithstanding the failure to have operable fire barriers meeting the fire endurance rating criteria specified by Section III.G of Appendix R, a plant is not necessarily unsafe to continue operation. To the contrary, fire watches are judged by the NRC to be adequate." "The goal of the NRC staff's Thermo-Lag Action Plan is directed towards restoring the functional capability of fire barriers as soon as practicable. There is not a time limit associated with the use of fire watches as a compensatory measure."

[3] See Generic Letter 91-18, Revision 1 "Information To Licensees Regarding NRC Inspection Manual Section On Resolution Of Degraded And Nonconforming Conditions"

In general, the application of the screening figures in section 2 may be confusing, and lead to differing interpretations. Many plant do not define their post-fire safe shutdown paths along strict electrical divisional bases, and may utilize equipment from several electrical divisions to make up a single shutdown path. This approach makes it difficult to create the distinction between "SSD Train A Function" and "SSD Train B Function" as described in the figures in Section 2. Based on the underlying plant design, there may be several ways of accomplishing a particular SSD function (ex., Reactor coolant makeup), or only one way (ex., shutdown cooling). These plant-specific details may make the application of the screening figures in Section 2 difficult and imprecise. It is recommended that additional explanation be provided regarding what constitutes a "SSD Function" as they are being evaluated in this section.

Section 5.0, Fire Protection Risk Significance Screening Methodology - Phase 2

Step 3: Qualitative Evaluation of Findings
Step 3 states "Therefore, in order to perform this step, the existing plant conditions as noted by the inspection finding are evaluated against the deterministic/qualitative evaluation guidance and degradations categorization criteria established in **IP XXX, Appendix H.**[4] The most recent version of the FPFI Draft Inspection Procedure available (Prairie Island version) does not contain this guidance. This appears to be an extremely important aspect in the NRC's development of risk significance determinations associated with fire protection SSC material condition. It is recommended that this information be made publicly available for comment. The criteria used to distinguish between "low", "medium" and "high" levels of degradation are of critical importance, and should be thoroughly explained and understood, such that they are not overly reliant on judgement and interpretation.

Step 5: Determination of Fire Ignition frequency
The data provided in Table 5.4, "Generic Ignition Frequencies, Plant Buildings or Rooms" in some cases is 2 to 3 times higher than fire ignition frequencies published in the EPRI FIVE Methodology.[5] The fire initiation frequencies published in the EPRI FIVE Methodology were considered to be conservative at the time, since during their development the NRC had directed that fire events not be screened out of the database based on lack of a flaming fire in many cases (i.e., smoking equipment was called a "fire"). The net affect of these differences in data may result in an overstatement of fire risk, or confusion in the application and interpretation of the SDP. In addition, these frequencies do not account for resultant damage probability.

[4] IP XXX, "Fire Protection Functional Inspection (FPFI)"

[5] EPRI TR-100370, "Fire-Induced Vulnerability Evaluation (FIVE)"

It is not clear what the technical basis is for the data and their relative ratios in Tables 5.1, 5.2, 5.3, and 5.6.

Table 5.7, Estimated Likelihood Rating for Initiating Event Occurrence During Degraded Period
Table 5.7 implies that a fire resulting in a reactor trip should be occurring at least annually in the U.S. industry. This does not appear to agree with industry experience. Also, it is not clear if this represents a fire causing a reactor trip directly, or a fire followed by a subsequent controlled shutdown, the implications of which are entirely different.

Step 9: Modifications Necessary To Add Impact of Spurious Actuations
This section is extremely difficult to interpret, and requires significant clarification. It appears to assign a weight of 10 (i.e., delta CDF is increased by a factor of 10) to the significance calculated in the SDP. This appears to be an overstatement of the risk of spurious actuations. For a spurious actuation to have meaningful risk impact on the safe shutdown capability, it must satisfy all of the following:

- Have some relationship to the Post-Fire Safe Shutdown strategy being relied upon in the fire area under consideration;
- Become exposed to the fire (located in the plume or hot gas layer);
- Have the spurious actuation occur (while the probability of hot shorts leading to spurious actuations is currently subject to debate, the probability of this occurring is clearly not 1.0);
- Go un-mitigated until an un-recoverable condition has resulted (this implies that either mitigation is not possible, the absence of "mitigating procedures" should not be the determining factor when a risk-based analysis is performed);
- The fire must also affect redundant equipment, such that it may not be capable of functioning in the place of the damaged component described above.

Each of these bullets has it's own probability of occurrence. These must all be considered to determine the risk significance of spurious actuations on the safe shutdown capability. Assigning a "Delta CDF" of a factor of 10 is clearly excessive (1.1 may be more appropriate).

Comments by Page

Page A4-4, Figure 4-1, There is not a clear basis for statements made regarding the use of compensatory measures being less effective than installed equipment or barriers. Nor is there any basis for the greater than 30 day criteria. This appears to back-fit new regulatory requirements that are more restrictive than current Technical

Specification requirements. In various SERs NRC has reviewed and already approved the conditions of Appendix R Safe Shutdown Program, which include statements that effectively equate that a firewatch as equivalent capability for the degraded equipment. However, in this screening process, the NRC is concluding that they are not equivalent. This is an inappropriate way to change requirements for compensatory measures by raising the standards on which they are measured. It should also be noted that DD-99-3 (February 1993) concluded that fire watches are an adequate replacement for installed equipment or barriers.

Page A4-14, Step 3, at the end of the first paragraph, the inspector is referred to "IP XXX, Appendix H", has this been issued ?

Page A4-16, Table 5.4, the fire initiation frequencies provided in Table 5.4 do not account for resultant damage probability.

Page A4-25, example 2B, the statement is made that the 1 hour fire barrier is restored to its full function condition, then sates that it is then assigned a low degradation. It is not clear nor reasonable to assume a fully functional barrier is degraded.

Page A4-25, Example 2B, contains the following statement in the last paragraph: "Addition of Greens may become a White." This is in direct conflict with enforcement guidance regarding aggregation of results.

Appendix 5, Significance Determination Process and Enforcement Review Panel

Page A5-4, Comparison with Non Pilot Enforcement Policy

It is not clear what the purpose of this comparison is or how the results will be used. The new process should not be determining a Finding color based on previous enforcement policy.

Shutdown SDP

This process, specifically the guideline tables, does not appear to be risk-informed. A statement regarding the increased risk during shutdown due to tech spec requirements is not consistent with the guideline tables. Availability and the functional approach outlined in NUMARC 91-06 should not be confused with tech spec inoperability. Potential confusion or backfit considerations might result through the use of this screening tool. The functional considerations may "require" additional equipment beyond that required by the tech specs.

Most of the items at worst would be a tech spec violation, most are only utility procedure non-compliances. Are procedure non-compliances with no real impact now subject to notices of violation? Many of the entries are the same in all modes. Suggest making an all modes list of these and put the mode specific items in those lists.

In the last paragraph of the introduction section, the BWR statements are not supported by shutdown PSA due to the longer response time available and the fact that not all alternate decay heat removal methods use the SRVs. In fact when flooded up if, as per their checklist, main steam line plugs are used, the SRVs are not available. The lists do not credit methods which plants use when RHR is out of service of for loss of normal DHR. Specifically, fuel pool cooling system and reactor water cleanup system are used in outages. They seem to only credit alternate shutdown cooling (their feed and bleed). They also do not seem to credit natural circulation either in PWRS or BWRs. The document should allow alternate methods of decay heat removal when flooded and allow all DHR to be turned of for a reasonable time (during vessel internals inspection, all DHR is out of service for 23 hrs/day for several days which represented a water temperature increase to about 140 degrees.

During hot shutdown, the required diesel generators could be related to required ECCS, DHR, etc or plant specific tech specs. Also, the AC items do not consciously acknowledge the >3 diesel plants. As written the guidelines would prevent 1e work at a PWR for a large portion of the outage and probably prevent the LOOP-LOCA tests at the beginning of a BWR outage.

The BWR cold shutdown, time to boil <2 hours, level <23ft and the BWR cold shutdown time to boil >2 hrs, level <23 ft lists appear to be identical, either differentiate them or combine them.

This document appears to be based on design bases and tech specs and not enough on defense in depth.

The Shutdown SDP process does not appear to be consistent with the philosophy of the "at power" SDP. Credit is given for non-tech spec systems and operator actions in the "at-power" SDP.

The expectation was that the NRC was codifying the practices that the utilities have been following in response to NUMARC 91-06. The 91-06 defense in depth philosophy credits non-tech spec or design basis systems and insures adequate defense in depth, for example using two fuel pool cooling systems and RWCU instead of RHR and feed and bleed, but having adequate defense in depth.

One of the iterations of the shutdown rule - the NRC allowed credit for time to boil as equivalent to another DHR system for defense in depth when flooded up.

Checklist items for RHR temperature instrumentation will have to be changed to a more general statement of indication of reactor water temperature if alternate DHR systems are credited.
The introduction used (defined) 'available' yet the check list used the term operable. Some sites will defeat all auto injection capability for ECCS per procedure for both personnel and safety reasons. When men are working in the cavity (as a lesson learned) they prevent inadvertently injected while the cavity is flooded. They voluntarily enter the action statement for no operable ECCS systems, and rely on manual action. This may not be acceptable per the check list.

The checklist seems to be too restrictive at times as to what can be credited for alternate DHR systems. Plants use main steam line plugs but installs and removes them when the cavity is flooded. The check list only seems to take credit for the plugs when cavity level is down. This would have impact on current methods equipment needed, dose, and ultimately critical path.

It also says to PM / test the cavity seals. Some plants have passive seals that are inspected on an infrequent basis. They do have a leak detection system. To increase the frequency to every outage prior to flood up would be a direct impact on critical path and would be a significant increase in dose.

Once cavities are flooded experience has shown that shutdown cooling can be removed for several hours and the heat-up seems to be limited to about 130 degrees, yet the check list says one loop of RHR operable and in operation. If this requirement is put in place then some invessel work and inspections would require off loading the core.

This check list may put restrictions on the sites schedule outages with little or no gain on risk reduction. List may need to be modified to be site specific, and a lot of exceptions and discussion may be required to show how each site is meeting the checklist requirements.

ENCLOSURE 5
Comments on the Pilot Program for the New Reactor Oversight Program
NRC Website

The NRC website for the new reactor oversight process consists of a number of separate pages that describe the new process, provide a status of the pilot program, show what's new on the website, and list frequently asked questions. In general, the website is organized to easily provide a public user with sufficient information to become familiar with the new process.

However, there are some parts of the website that may cause confusion for an infrequent user. For example, on the introductory web page http://www.nrc.gov/NRR/OVERSIGHT/index.html) there is a link to "Frequently Asked Questions" (FAQ) which goes to a page of "General Overview Questions." If you go to the "Program Overview" page the FAQ section has two additional FAQ categories, "Philadelphia Workshop" and "Emergency Preparedness Cornerstone Questions."
Recommendation: The FAQ web page should have all the FAQs or links to them. Another example is the "Program Information" section of the overview web page still does not have links to descriptions of the baseline inspection program, action matrix, and the significance determination process.

In addition, the question answers provided should be reviewed to ensure they are consistent with NEI 99-02 Draft Rev. D. (This document contains answers that have been accepted by NRC. Examples were noted where answers did not agree with NEI 99-02.

The Revised Reactor Oversight Process (Pilot) Plant Assessment and Results web page is a very good public display of the current status of the NRC assessment of each plant. The "Performance Summary" page with the links to all the supporting data provides any interested user with considerable information concerning the assessment of each reactor plant. The access to all this information, however, provides the opportunity to undermine the credibility of the summary web page if the information is not consistent at the various locations, or is not clearly explained.

A review of the pilot plant performance summary pages provided the following examples of inconsistency.

- There were six examples of findings listed on the web page that were in different cornerstones than listed in the Inspection Report.

- There were six examples where findings were listed under the wrong reactor unit for multiple unit sites.

- There were four examples of findings that were listed in an Inspection Report and not on the web page for that reactor unit or vice versa.

- There were two examples of finding significance on the web page listed as N/A when the finding was listed as "no color" in the Inspection Report.

A review of the pilot plant performance summary pages provided the following examples where the information was unclear or confusing.

- There were ten examples of findings where the "Item Type" was listed as "FIN Finding." There is no explanation as to what a "FIN Finding" is.

- There were findings listed where the discussion, either on the web page or in the Inspection Report, was not sufficient to enable the reader to understand why the issue was a finding. Examples include:
  - A finding concerning corrective action program effectiveness; the only adverse comment is that the program was complicated.

  - A finding concerning the Offsite Dose Calculation Manual (ODCM); the inspection report said this issue "did not constitute a violation of regulatory requirements." and was "of low risk significance".

There was one example in the Performance Indicators where the licensee made a comment about performance data that was not available on the web site.

There are some minor changes recommended that would help web site users more clearly use or more clearly understand the information provided:

- On the plant performance summary pages, a code letter should be added to the indicator blocks to show what the color is when the page is printed on a black and white printer, similar to how it is done now on the inspection findings page.

- On the web page "Description of Cornerstones and Performance Indicators" , there should be a discussion of and a link to NEI 99-02, so that if a person is interested, they can go to the reference manual and get the details of how PI's are calculated. Also, it might help to provide an explanatory paragraph on why some PI's don't have yellow or red thresholds.