UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

February 16, 2001

Mrs. Kay Drey
515 West Point Avenue
University City, MO 63130

SUBJECT: CALLAWAY PLANT REFUELING OUTAGE 10 AND NRC INSPECTION 50-483/2000-17

Dear Mrs. Drey:

We have completed our review of the questions included in your December 27, 2000, letter. In some cases, we concluded that we had insufficient documented information to answer your questions. For example, some of the information you requested was not collected under the Revised Reactor Oversight Program, or was not documented because it was not needed to make a determination about licensee performance relative to the issues documented in NRC Inspection Report 50-483/2000-17. Nevertheless, our answers to your questions are provided in Enclosure 1 to this letter.

As you requested, a copy of NRC Inspection Procedure 71121.02, "ALARA Planning and Controls," is attached to the enclosure. Additional background information has also been attached, and NRC inspection manual chapters and inspection procedures are available from our web site at the address provided in the enclosure.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the ADAMS Public Library component on the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

As I stated in my January 29, 2001, letter to you, we appreciate your interest in these matters.

Sincerely,

for Gail M. Good, Chief
Plant Support Branch
Division of Reactor Safety

Docket: 50-483
License: NPF-30

Enclosure:
Questions and Responses

Attachments:
1. Inspection Procedure 7112102
2. Reactor Oversight Process Description
3. Final Significance Determination and Notice of Violation (NRC Inspection Report 50-483/00-17), dated January 9, 2001
4. Licensee Appeal, dated February 7, 2001
5. Manual Chapter 0305, “Operating Reactor Assessment Program”
6. Manual Chapter 0609, Attachment 0609.02
7. Manual Chapter 0610*, “Power Reactor Inspection Reports”
8. Inspection Report 50-483/00-17, dated October 4, 2000

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ENCLOSURE

Question 1:

In the middle of the refueling outage, AmerenUE decided not to revise its original occupational radiation dose projections even though the projected levels were already being exceeded. According to UE's August 21, 2000, letter to the NRC, UE's reasoning was "to avoid worker let down as work progressed toward achieving the original goal." Would you please tell me how the NRC interprets "worker let down?" (ULNRC-4298, p.3)

NRC Response:

The NRC interpreted "worker letdown" as meaning a decrease in worker morale, caused by increasing the outage dose projections (i.e., a criticism or disincentive).

Question 2:

May I please have a copy of the NRC Inspection Procedure 71121.02 regarding ALARA Planning and Controls? Would you please tell me how to obtain a copy of the "Occupational Radiation Safety Cornerstone?" And, is it possible to access the Callaway "Suggestion-Occurrence-Solution" reports online?

NRC Response:

A copy of NRC Inspection Procedure 71121.02, "ALARA Planning and Controls," is enclosed (see Attachment 1). Additional NRC manual chapters and inspection procedures are available on the NRC web site at http://www.nrc.gov/NRC/IM/index.html.

The Occupational Radiation Safety Cornerstone is a concept, not a document. It is one of seven cornerstones in the NRC's Reactor Oversight Program that support the three Strategic Performance Areas (Reactor Safety, Radiation Safety, and Safeguards). Performance information is summarized for each plant and sorted by the seven cornerstones of safety. The objective of the Occupational Radiation Safety Cornerstone is to ensure adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. The Reactor Oversight Program cornerstones are discussed in Attachment 2 and on the NRC web site at http://www.gov/NRR/OVERSIGHT/ROP/description.html#cornerstones.

The licensee's Occurrence-Solution reports are not accessible online.

Question 3:

Has the NRC Staff made its decision as yet regarding whether to classify UE's high collective doses during Refueling Outage 10 as "white" or as the lesser category, "green?" That is, have you decided whether increased NRC oversight is warranted, or not?

NRC Response:

The NRC concluded that the inspection findings discussed in NRC Inspection 50-483/2000-17 were appropriately characterized as three White findings. The licensee was notified of the final significance determination in a letter date January 9, 2001 (see Attachment 3). We have also enclosed a copy of Callaway's Appeal of the Final Significance Determination dated February 7, 2001, which is in response to the above letter (see Attachment 4). Accordingly, the NRC's regulatory response will follow the guidance within the Action Matrix in NRC Inspection Manual Chapter 0305, "Operating Reactor Assessment Program" (see Attachment 5).

Question 4:

Was the confusion about the term, "job" resolved as a result of the November 9 conference? I remain uncertain as to how or whether that word relates to the "collective radiation dose."

NRC Response:

The term "job" relates to the level at which a licensee implements ALARA planning and controls. There were differing views between the NRC and the licensee as to how much work should be considered in an individual job and, consequently, how much dose was accrued by the job. After reviewing supplemental information in the licensee's November 16, 2000, letter, the NRC concluded that the licensee implemented ALARA planning and controls at the radiation work permit level rather than at the work authorizing document level (refer to NRC's final significance determination letter dated January 9, 2001 [see Attachment 3]). Therefore, in determining the significance of the inspection finding, we concluded that it was appropriate to equate the dose accrued by a job with the dose accrued by the radiation work permit.

Each job dose contributes to the collective radiation dose. However, the significance attributed to a finding by the Occupational Radiation Safety Significance Determination Process is dependent on the amount of dose accrued by the associated job, not directly to the collective radiation dose. (The collective radiation dose history, in the form of the 3-year rolling average, collective dose, is considered before the Occupational Radiation Safety Significance Determination Process is used.) Refer also to NRC Inspection Manual Chapter 0609, Attachment 02, "Initial Assessment of Inspection Observations for Significance Determination Process Entry," http://www.nrc.gov/NRC/IM/0609-02.html . (see Attachment 6).

Question 5:

At the November 9 regulatory conference UE apparently blamed its difficulties during the refueling outage in part on a "lack of incentive to get experienced personnel to return for outages" and on "worker inexperience." If you asked UE at the conference why it had employed inexperienced workers for the refueling, did you receive a satisfactory response, and if so, would you please tell me what it was? (NRC letter to UE, Nov. 15; Enclosure 2 - "Licensee Presentation," pp. 10, 12)

NRC Response:

We understand that the licensee's agreement with its primary contractor lacked monetary incentives to induce experienced contract workers to return to the site for Refueling Outage 10. Without this incentive, many experienced workers went to other nuclear sites conducting concurrent refueling outages that did offer incentives.

Question 6:

What foreign object(s) was (were) retrieved during the outage? (NRC IR #2000-17, p.8)

NRC Response:

NRC Inspection Report 50-483/00-17 referenced Radiation Work Permit 99-53022, foreign object search and retrieval. Foreign objects retrieved under this radiation work permit, according to information provided by the licensee, were parts of a 3-inch diameter steel gasket ring inside the secondary side of Steam Generator D. The origin of the parts could not be determined.

Question 7:

Did the NRC Staff know that UE was planning to replace its steam generators in the year 2005 when it decided to approve the experimental steam generator tube electrosleeving during the October 1999 refueling? (As no doubt expected, the steam generator retrofitting did indeed cause many workers to be exposed to high levels of radiation because of the notoriously high radiation fields that exist in, under, and near all nuclear steam generators.)

NRC Response:

The NRC knew that the licensee was planning to replace its steam generators about 2005 when it approved the electrosleeve amendment to the Callaway Technical Specifications. The licensee stated that it planned to use the electrosleeve process to avoid plugging steam generator tubes (and possibly having at some point to reduce power) before it was prepared to replace the steam generators. The licensee projected the replacement would be performed in 2005. Therefore, when the electrosleeve amendment was approved, the staff knew that the electrosleeve process would be used in several refueling outages before the steam generator replacement.

Question 8:

(a) Would you please tell me how many individual workers participated in the steam generator activities - that is, including the removal and installation of the steam generator manway covers and inserts [8.543 person-rem collective dose]; the eddy current/robotic plugging/stabilizing/ electrosleeving [57.659]; and the health physics support for primary and secondary steam generator activities [5.641]? (NRC IR #2000-17, Encl. p.4); and (b) What was the highest accumulated exposure of an individual worker during the steam generator job(s), and what was his or her craft?"

NRC Response:

The specific number of workers in each of these activities was not documented because the information was not needed for the NRC to make a determination about the licensee's program. The NRC reviewed dose information to determine whether: individuals received radiation doses in excess of the regulatory limits of 10 CFR 20.1201, individual jobs exceeded 5 person-rems, individual jobs exceeded their projected dose by more than 50 percent, and the 3-year rolling average, collective dose exceeded 135 person-rems.

The highest individual cumulative dose of an individual involved in steam generator work was below the regulatory limits of 10 CFR 20.1201 and, therefore, was not documented in accordance with the guidance of NRC Manual Chapter 0610*, "Power Reactor Inspection Reports" (see Attachment 7).

See http://www.nrc.gov/NRC/IM/0610star.html or http://www.nrc.gov/NRC/IM/index.html.

Question 9:

As I understand it, a severe reactor fuel axial offset anomaly was first detected at Callaway during the plant's fifth operating year (May 1989-September 1990) - that is, an anomalous shifting between levels of power and temperature in the upper and lower portions of the fuel core. This anomaly potentially can generate such significant risks as local power peaking of the fuel rods, a reduction in the shutdown margin, and anomalous control rod positioning and fuel rod cladding temperatures. (a) Does Callaway still hold the record as having experienced the largest axial offset known (-15 percent, according to the NRC's IR # 1997-19, p.4)? (b). Was the axial offset anomaly addressed during the tenth refueling outage, or if not, when was this problem resolved at Callaway?

NRC Response:

The NRC does not have current information on Callaway's axial offset anomaly relative to other sites. However, a memorandum, dated January 26, 2001 (ADAMS Accession No. ML010300078), from the NRC senior project manager for Callaway may address part of your question. It states, in part:

> **The axial offset at the beginning of Cycle 10 was more negative than predicted and it was apparent that the phenomenon was not understood and the analytical tools being used needed to be adjusted. Throughout Cycle 10, periodic updates were provided and the licensee kept SRXB [Reactor System Branch] informed as to progress including actions planned to eliminate the problem for Cycle 11 operation. For the first part of Cycle 11, the agreement between predicted and measured axial offset was quite good. However, at a burnup of about 7 GWD/MTU, [giga watt days/metric tons uranium] the axial offset anomaly was again observed. Since that time, the licensee has taken several steps to mitigate the problem, including: performing weekly flux maps, developing control rod exercise**

guidance to minimize crud releases, updating core follow models, improving sampling techniques, and controlling chemistry tighter.

After many months, the axial offset is improving and no shutdown margin challenge is anticipated for the remainder of Cycle 11. In a conference call on November 16, 2000, the licensee described the activities that are underway to eliminate the axial offset anomaly for Cycle 12 and the actions to be taken during Cycle 12. The licensee has been aggressively working on this problem and, based on the information provided and our understanding of the phenomena of axial offset anomaly, no further work is needed on this TAC [technical assignment control].

Question 10:

(a) Would you please describe the composition of the current Callaway fuel core, following the 10th refueling --- that is, the number of fuel assemblies of each U-235 enrichment level; the number of control rod assemblies and the number of control rods in each; and the number of burnable poison rods used during and subsequent to the 10th refueling? (I am particularly interested in learning if any of the fuel is enriched at greater than 4.6%, which I believe is the highest enrichment level I have read about at Callaway.)

(b) Has UE, Westinghouse or the NRC determined whether the higher enriched fuel, in a batch of split-enrichment fuel, can contribute to axial offset, power peaking or crud buildup?

NRC Response:

(a) The following information is based on data taken from the licensee's Updated Final Safety Analysis Report, Revision OL-11, May 2000. The tabulated data for the current fuel cycle provides the number of fuel assemblies for each U-235 enrichment level.

CURRENT CYCLE FEED ASSEMBLY FUEL		
Region	Number or Assemblies	Enrichment (w/o U235)
12A	56	3.80
12B	40	4.20
12C	1	2.80
13A	36	4.2
13B	60	4.6

The number of full-length control rod assembly clusters in the reactor core is currently 53 with 24 adsorber rods per cluster. As determined from the above data table, there is no fuel currently in the reactor core with enrichment greater than 4.6 percent. In addition, it was determined that at the beginning of Operational Cycle 10, 576 wet annular burnable adsorbers were loaded in the reactor core. During the 10the refueling, these burnable

adsorbers were removed from the reactor core, and 368 wet annular burnable adsorbers were loaded into the reactor core for Operational Cycle 11.

(b) The NRC has no information regarding this question. Either the information was not collected under the Reactor Oversight Program, or the associated issue was determined to have no safety significance and, therefore, was not documented. See NRC Manual Chapter 0610*, "Power Reactor Inspection Reports," Appendix B, "Thresholds for Documentation" (Attachment 7).

The licensee may be able to provide this information.

Question 11:

Regarding the crud removal activities during the refueling outage: to what extent does the NRC Staff estimate the high collective worker doses were attributable to the buildup of crud on the fuel rods and elsewhere in the primary cooling system, in the secondary cooling system, and in other out-of-core locations? (An additional question about which I'm just curious: does the nuclear industry still call the crud in the secondary cooling system "the green grunge?")
*

NRC Response:

The NRC made no attempt to quantify the percentage of collective dose caused by the buildup of crud on the fuel rods, in the reactor primary system, the reactor secondary system, or other out-of-the-core locations. NRC regulations apply to radiation dose regardless of the source. The licensee or other industry organizations, such as the Electric Power Research Institute (EPRI), may be able to provide you with this information.

The NRC has no information about the use of the term, "the green grunge."

Question 12:

To what extent was the crud on the reload fuel rods/assemblies removed by ultrasound, before placing the fuel in the spent fuel pool for storage during the outage, or after removing it from the pool prior to replacing it in the reactor vessel? Was that job performed by robots? If not, what was the estimated collective dose of the decontamination workers?

NRC Response:

The NRC made no attempt to determine the extent to which the crud on the reloaded fuel rods and assemblies was removed by ultrasound or by what means it may have been accomplished. This information was not needed for us to evaluate licensee performance using the Reactor Oversight Program.

Question 13:

If UE monitored the cobalt-58 and -60, and other dissolved or dislodged corrosion products that accumulated in the reactor coolant during the refueling outage --- to assess the efficacy and efficiency of the crud removal activities --- would you please tell me the total number of curies calculated?

NRC Response:

We did not collect or document this information because it was not needed for us to evaluate licensee performance using the Reactor Oversight Program.

Question 14:

Were chelating agents used to dissolve the crud during the refueling outage? Are chelating agents or other solvents allowed to be added at Callaway as a part of the primary or secondary coolant water chemistry while the reactor is on line?

NRC Response:

The NRC did not determine if the licensee used chelating agents to dissolve the crud. This information was not needed for us to evaluate licensee performance using the Reactor Oversight Program. Typically, hydrogen peroxide (not a chelating agent) is added to initiate a crud burst. See the response to Question 15.

Question 15:

(a) Do you know if more than one reactor coolant system crud burst occurred during the refueling outage? (Apparently at least one occurred while the workers were installing the scaffolding in the Reactor Building, as cited during UE's presentation at the November 9 regulatory conference. Encl. 2, p.12) (b) Could you please briefly describe a crud burst? (c) To what extent was(were) the crud burst(s) responsible for the 46.345 person-rem dose experienced by the scaffolding workers? (d) How many workers were exposed during the scaffolding activities? (e) Were members of crafts other than carpenters included in that total collective personnel dose? (f) What was the highest accumulated dose of an individual worker during the scaffolding job(s), and what was his or her craft?

NRC Response:

The licensee initiated a planned crud burst during reactor shutdown. The NRC has no information about additional crud bursts.

A crud burst is the resuspension of activated corrosion products within the reactor coolant system. A crud burst may be caused by a chemical, thermal, or hydraulic shock. Examples of some effective measures to reduce post-shutdown dose rates at pressurized water reactors through planned crud bursts are listed in NRC Inspection Procedure 79702, "Control and Monitoring of Radiological Source Term." These are:

- **Early boration at shutdown to promote corrosion product solubility and aid in the release of corrosion products,**

- **Slow plant cooldown rates with specific temperature hold points to maintain solubility of corrosion products, thus aiding in their removal via filtration and ion exchange,**

- **Use of hydrogen peroxide to create an oxygenation crud burst, liberating corrosion products early on in the shutdown clean-up process, and**

- **Use of maximum primary clean-up purification flow to enhance corrosion product removal [via filtration and ion exchange].**

The NRC made no attempt to quantify the percentage of collective dose that was received by the scaffolding workers who worked near the reactor coolant system when the crud burst was conducted. The NRC reviews personnel dose, regardless of the source, and compares the dose to regulatory requirements and significance determination process screening questions and logic steps.

The specific number of workers in each of these activities and their associated crafts were not documented because the information was not needed to access licensee performance using the Reactor Oversight Program. See the response to Question 8.

As in Question 8, the highest individual cumulative dose of an individual involved in scaffolding work was below the regulatory limits of 10 CFR 20.1201.

Question 16:

And one final question, -- about valves, based on UE's Oct. 9, 2000, letter to the NRC (ULNRC-4321): According to the Callaway Licensee Event Report #2000-5, dated September 7, 2000, UE discovered in September that surveillance testing of the automatic isolation valves BNHV8812 A and B (at the interlock between the pumps of the Residual Heat Removal containment sump and the Refueling Water Storage Tank) had not been performed every 18 months as required by the Callaway Technical Specifications. In fact, the valve surveillances had been overlooked since the plant first became operable, fifteen years ago.

Apparently UE had performed maintenance testing of a portion of the circuitry for Valve B during the Refuel 10 shutdown, in October 1999, and of the rest of the circuitry for Valve B in August 2000. But since the valve surveillances can only be performed when the plant is not operating at power, would you please tell me when UE expects to test the Valve A circuitry, and the rest of the mandatory tests (thus far overlooked) of both Valves A and B?

As stated in Licensee Event Report 483/2000-005, required surveillance testing was completed for Valve BNHV8812B on August 31, 2000. Required surveillance testing of Valve BNHV8812A will be completed as required by Technical Specification Amendment 140, issued on October 6, 2000. This amendment added the following note under Surveillance Requirement 3.5.2.5: "Verification of the automatic closure function of

BNHV8812A shall be performed prior to startup from the first shutdown to MODE 5 occurring after September 8, 2000, but no later than June 1, 2001." We understand that the required testing is scheduled to be performed during the Spring 2001 refueling outage.

ATTACHMENT 1

Issue Date: 04/03/00

Attachment 71121.02

Table of Contents

Inspectable Area: ALARA Planning And Controls

Cornerstone: Occupational Radiation Safety

Inspection Basis: This inspectable area verifies aspects of the Occupational Radiation Safety cornerstone for which there are no indicators to measure performance. The stochastic risk effect of exposure is based on the linear non-threshold exposure model. Increasing individual or collective exposures equates to increased risk of cancer or genetic effects.

Level of Effort: Inspect Biennial

71121.02-01 Inspection Objective

01.01

To assess performance with respect to maintaining individual and collective radiation exposures as low as is reasonably achievable. This inspection will determine whether the licensee has an adequate program, including administrative, operational, and engineering controls, to maintain occupational exposure ALARA.

71121.02-02 Inspection Requirements

Note: This inspection may be performed during plant operations, with respect to on-line maintenance, when the ALARA review time frame is compressed, or this inspection may be performed during outage conditions. Short ALARA inspections may also be considered approximately 2 months prior to and 2 months following a significant maintenance or refueling outage to evaluate planning and final performance results, respectively.

02.01 Inspection Planning

a. Review pertinent information regarding plant collective exposure history, current exposure trends, and ongoing or planned activities in order to assess current performance and exposure challenges. The overall collective exposure performance may be utilized to provide a perspective of significance for inspection finding assessment.

b. Review outage or online maintenance work scheduled during the inspection period and associated exposure estimates or previous job history data. Select 5-10 work activities which are likely to result in the highest personnel collective exposures.

c. Review the plant's collective exposure data for the previous year and for the three- year rolling average and compare this with the comparable numbers for all BWRs or PWRs.(1) This will provide a sense of the significance of collective exposure at the facility.

d. Using available data, determine the site specific trends in collective exposures and source-term(2) (average contact dose rate with reactor coolant piping) measurements.

e. Review site specific procedures associated with maintaining occupational exposures ALARA. Include a review of processes used to estimate and track job specific exposures.

The site specific and industry-wide bench-marking provides a relative perspective of "reasonableness" and should be considered when assessing and documenting most ALARA inspection findings.

02.02 Job Site Inspections and ALARA Control

NOTE: Job site inspection activities may be combined with Section 02.04 of the Access Control to Radiologically Significant Areas procedure.

a. Based on scheduled work activities and associated exposure estimates, select about 5 high exposure or high radiation area active job locations and evaluate the licensee's use of ALARA controls by performing the following:

1. Survey (directly or utilizing the services of an RP technician) the work area and identify the source location(s) and low dose area(s). Dose rate gradients are often indicative of sources that are not effectively shielded. These areas may be further investigated to determine the basis for the as-found source configuration in applicable ALARA reviews.
2. Evaluate the licensee's use of engineering controls to achieve dose reductions. Utilize ALARA reviews as criteria for this evaluation.
3. Determine if workers are utilizing the low dose waiting areas and are effective in maintaining their doses ALARA (e.g., do they remain in the work area or move to the low dose waiting area when subjected to temporary work delays).
4. Determine if workers receive appropriate on-the-job supervision to ensure the ALARA requirements were met. Determine if the first-line job supervisor ensures the job is conducted in a dose efficient manner (e.g., work crew size minimized, workers properly trained, proper tools and equipment available at start of job, etc.).

b. Review individual exposures of selected work groups. Determine the reasons for any significant exposure variations which may exist among workers.

02.03 Source-Term Reduction and Control

a. Utilizing licensee records, determine the historical trends and current status of tracked plant source terms.

Consider utilizing a survey instrument to walk down selected accessible areas of the station to verify the accuracy and completeness of the licensee's source tracking program. During plant tours investigate any sources that may affect collective exposures and are not tracked by the licensee.

b. Using licensee records, determine whether the overall plant source-term is increasing, stable, or declining. Determine if the licensee has developed an understanding of the plant source-term, including knowledge of input mechanisms to reduce the source term. Determine whether the licensee has a source-term control strategy in place. This should include a cobalt reduction strategy and shutdown ramping and operating chemistry plan (designed to minimize the source-term external to the core) as a minimum. If not, look for reasonable justifications for not pursuing such an exposure reduction initiative.

c. If the licensee has a source-term control strategy in place, determine if specific sources have been identified by the licensee for exposure reduction actions and what priorities the licensee has established for implementation of these actions. Determine what results have been achieved against these priorities since the last refueling cycle. During the current 12 month assessment period, determine whether source reduction evaluations have been made and actions have been taken to reduce the overall source-term compared to the previous year.

02.04 Radiation Worker Performance.

During on-site inspection, observe radiation worker and RP technician performance during high dose rate or high exposure jobs and determine if workers demonstrate the ALARA philosophy in practice (e.g., are workers familiar with the job scope and tools to be used, are workers utilizing ALARA low dose waiting areas), whether there are any procedure compliance issues. Also, observe radiation worker performance to determine whether the training/skill level is sufficient with respect to the radiological hazards and the work involved.

02.05 Radiological Work Planning.

Obtain from the licensee a list of jobs ranked by estimated exposure that are in progress or that have been completed during the last outage (the most recent outage ALARA report may be a good source for obtaining dose estimates and actual doses for previously completed jobs). Evaluate the exposure estimates and exposure performance data based on plant job history or relevant industry performance data (as available). Identify any jobs where the actual job exposure was greater than the estimated job exposure by more than 50% and the actual job exposure exceeded 5 person-rem.

a. If more than 5 jobs meeting these criteria were identified, select the 5 jobs of highest exposure significance.

b. Review the ALARA job evaluations, exposure estimates, and exposure mitigation requirements. Compare these ALARA plans with results achieved (dose rate reductions, man-hours used). If results are consistently different, find out why.

c. Evaluate the interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling and engineering groups for interface problems or missing program elements.

d. Review the integration of ALARA requirements into work procedure and RWP documents.

e. Evaluate the accuracy of person-hour estimating provided by maintenance planning to the radiation protection group and person-hour tracking provided by radiation protection.

f. Evaluate the radiation protection group generated shielding requests with respect to dose rate reduction problem definition and assigning value (dose savings or dollars). Evaluate engineering shielding responses for follow through.

g. Determine if jobs are scheduled to consider the benefits of dose rate reduction activities to include: water shielding from pipe filled conditions, job scheduling, and shielding and scaffolding installation and removal activities.

h. Determine if post-job reviews were conducted and if identified problems were entered into the licensee's corrective action program.

02.06 Verification of Exposure Estimate Goal and Exposure Tracking Systems

NOTE: The significance of ALARA findings will often depend on reasonably accurate exposure estimates. Reasonable implies that they be based on good assumptions and correct calculations with some flexibility given with regard to expected variability due to the limits of forecasting.

a. Review the assumptions and basis for the current annual exposure estimate and annual exposure goal. Review applicable procedures to determine the methodology for estimating job-specific exposures. Evaluate both dose rate and man-hour estimates for accuracy. Look for bottom-up (aggregation of individual job estimates) exposure estimates corroborated by top-down (past outage dose/day times days) estimating methods. Use of past outage experience combined with additional industry experience can provide a reasonable exposure estimate approach. If exposure estimates appear questionable, use site-specific past experience as the primary standard of comparison and utilize industry data (as available) of actual job exposure data as a secondary standard of comparison to determine the reasonableness of licensee exposure estimates.

b. Review actual exposure results versus initial exposure estimates. For the same jobs, review and compare the estimated and actual dose rates and man-hours expended. Determine if dose rate estimating and man-hour estimating are reasonably accurate when compared to actual results.

c. Review the licensee's exposure tracking system. Determine whether the level of exposure tracking detail, exposure report timeliness and exposure report distribution is sufficient to support control of collective exposures. For example, do RWPs cover too many jobs to allow job specific exposure trends to be detected and controlled? During the conduct of exposure significant maintenance work, look for evidence that licensee management was aware of the exposure status of the work and would intervene when exposure trends increase beyond exposure estimates.

02.07 <u>Declared Pregnant Workers</u>.

Determine if there have been any declared pregnant workers during the current assessment period. Review the exposure results and monitoring controls employed by the licensee with respect to requirements.

02.08 <u>Problem Identification and Resolutions</u>

a. Review audits and self-assessments for the ALARA program. Review dose significant post-job reviews and post-outage ALARA report critiques of exposure performance. Determine if identified problems are properly characterized, prioritized, entered into the corrective action program, and resolved in an expeditious manner.

b. Identify any jobs that resulted in collective exposures > 5 rem and where the actual collective exposure exceeded the initial exposure estimate by > 50%. Develop an inspection finding for those jobs identified. Examine the causes for exceeding the exposure estimate.

c. Review post-job reviews and post-outage ALARA report critiques for about 5 of the most dose significant jobs that have occurred since the last inspection in this area. 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. Initial problem identification, characterization, and tracking.
2. Disposition of operability/reportability issues.
3. Evaluation of safety significance/risk and priority for resolution.
4. Identification of repetitive problems.
5. Identification of contributing causes.
6. Identification and implementation of long-term corrective actions.
7. Resolution of non-cited violations (NCVs) tracked in corrective action system(s).
8. Implementation/consideration of risk significant operational experience feedback.

Emphasis should be placed on ensuring that problems are identified, characterized, prioritized, entered into a corrective action, and resolved.

d. For repetitive deficiencies or significant individual deficiencies in problem identification and resolution identified above, determine if the licensee's self-assessment activities are also identifying and addressing these deficiencies.

71121.02-03 Inspection Guidance

03.01 **No inspection guidance provided.**

03.02 **Job Site Inspections and ALARA Control**

Performing surveys of high collective dose job locations can provide a means to evaluate the results of the licensee's ALARA efforts. If the sources are shielded to essentially background levels, then no significant dose gradients should be detected and review of the licensee's shielding efforts in this area may not be necessary.

03.03 **Source Term Reduction and Control**

If any radiation sources that may affect collective exposures and are not tracked by the licensee have been identified, determine how long the condition has existed, if postings and radiation surveys have been deficient, whether any unplanned exposures have occurred or were likely to occur, and whether the licensee has entered this finding into their corrective action program.

If a licensee identified radiation source is old (greater than 1 year), determine how long the condition has existed. If this source may have resulted in unnecessary exposures (such as during an outage), determine how much exposure has resulted from (or was likely to have resulted from) the source and compare those results with the licensee's exposure evaluation assessment to encompass the extended time period.

If actions taken to reduce the source term have been ineffective, determine if follow up evaluations and additional actions have been planned. If not, look for additional examples to establish whether there is a pattern.

03.04 **No inspection guidance provided.**

03.05 **Radiological Work Planning**

Were there multiple examples where a job exceeded 5 person-rem and where this total job exposure also exceeded the projected job exposure by more than 50%? If so, were there similarities in the breakdowns in work planning which resulted in each of these jobs exceeding their job estimates by greater than 50%? Determine if emergent work conditions may have been responsible for any of these jobs exceeding the projected job exposures by more than 50%. A job may have benefitted from proper ALARA radiological work planning, yet overshot it's projected job exposure due to unplanned, unexpected emergent work. If these jobs were entered into the licensee's corrective action program, were appropriate licensee organizations held accountable for these breakdowns in work planning? The ALARA rule in 10 CFR 20 does not require every ALARA effort to demonstrate optimized exposure performance.

03.06 through 03.08 No inspection guidance provided.

71121.02-04 Resource Estimate

For planning purposes, it is estimated to take, on average, 120 hours biennially to perform the requirements of this attachment.

END

1. NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities". This data is available on the NRC external Web page.

2. EPRI TR-108737 (Dec 1998), "BWR Iron Control Monitoring Interim Report" [average BWR source-term based on this report is 220 mrem/hr]. EPRI TR-107566 (Feb 1997), "Evaluation of PWR Radiation Fields: 1991-1996" [average PWR source-term based on this report is 100 mrem/hr]. Source-term as defined by EPRI means average contact dose rate with the vertical recirculation piping (for BWRs) and with the crossover loop elbow near the reactor coolant pump piping (SRMP pt C5) for PWRs.

ATTACHMENT 2

Reactor Oversight Process
Process Description

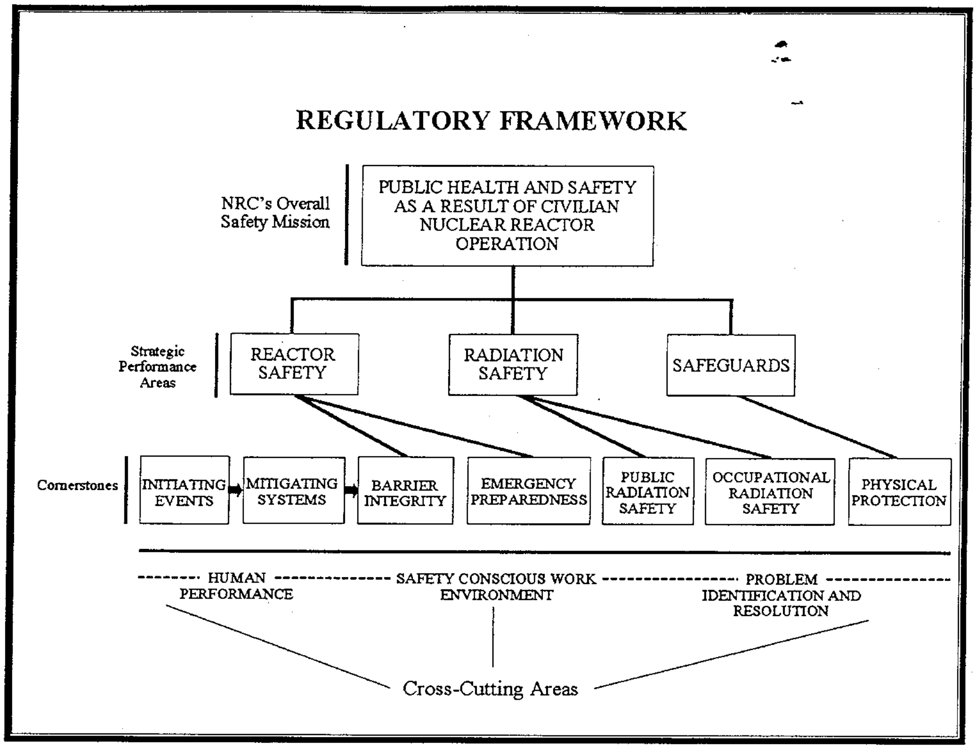
A description of the reactor oversight process is provided below. You may click on any item in Contents to go directly to it. You may also want to download or print the entire description -- it is approximately 10 pages long. Thank you for your interest in the NRC Reactor Oversight Process!

Contents

Regulatory Framework

The regulatory framework for reactor oversight is shown in the diagram below. It is a risk-informed, tiered approach to ensuring plant safety. There are three key strategic performance areas: reactor safety, radiation safety, and safeguards. Within each strategic performance area are cornerstones that reflect the essential safety aspects of facility operation. Satisfactory licensee performance in the cornerstones provides reasonable assurance of safe facility operation and that the NRC's safety mission is being accomplished.

Within this framework, the NRC's operating reactor oversight process provides a means to collect information about licensee performance, assess the information for its safety significance, and provide for appropriate licensee and NRC response. Because there are many aspects of facility operation and maintenance, the NRC inspects utility programs and processes on a risk-informed sampling basis to obtain representative information.



The Cornerstones of Safe Operation

The new reactor oversight program is, of course, anchored in the NRC's mission to ensure public health and safety in the operation of commercial power plants. That will always remain the agency's overarching responsibility.

The objective is to monitor performance in three broad areas -- reactor safety (avoiding accidents and reducing the consequences of accidents if they occur); radiation safety for both plant workers and the public during routine operations; and protection of the plant against sabotage or other security threats.

To measure plant performance, the oversight program focuses seven on specific "cornerstones" which support the safety of plant operations in the three broad strategic areas.

Initiating Events - This cornerstone focuses on operations and events at a nuclear plant that could lead to a possible accident, if plant safety systems did not intervene. These events could include equipment failures leading to a plant shutdown, shutdowns with unexpected complications, or large changes in the plant's power output.

Mitigating Systems - This cornerstone measures the function of safety systems designed to prevent an accident or reduce the consequences of a possible accident. The equipment is checked by periodic testing and through actual performance.

Barrier Integrity - There are three important barriers between the highly radioactive materials in fuel within the reactor and the public and the environment outside the plant. These barriers are the sealed rods containing the fuel pellets, the heavy steel reactor vessel and associated piping, and the reinforced concrete containment building surrounding the reactor. The integrity of the fuel rods, the vessel, and the piping is continuously checked for leakage, while the ability of the containment to prevent leakage is measured on a regular basis.

Emergency Preparedness - Each nuclear plant is required to have comprehensive emergency plans to respond to a possible accident. This cornerstone measures the effectiveness of the plant staff in carrying out its emergency plans. Such emergency plans are tested every two years during emergency exercises involving the plant staff and local, state, and, in some cases, federal agencies.

Occupational Radiation Safety - NRC regulations set a limit on radiation doses received by plant workers, and this cornerstone monitors the effectiveness of the plant's program to control and minimize those doses.

Public Radiation Safety - This cornerstone measures the procedures and systems designed to minimize radioactive releases from a nuclear plant during normal operations and to keep those releases within federal limits.

Physical Protection- Nuclear plants are required to have well-trained security personnel and a variety of protective systems to guard vital plant equipment, as well as programs to assure that employees are constantly fit for duty through drug and alcohol testing. This cornerstone measures the effectiveness of the security and fitness-for-duty programs.

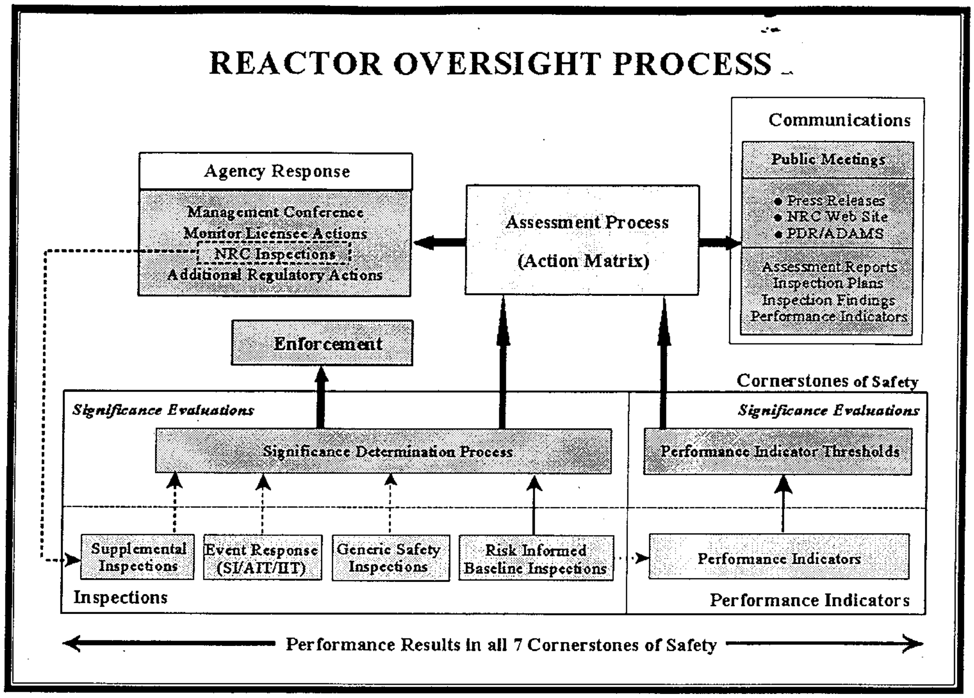
In addition to the cornerstones, the reactor oversight program features three "cross-cutting" elements, so named because they affect and are therefore part of each of the cornerstones:

- **Human performance**
- **Management attention to safety and workers' ability to raise safety issues** (The so-called "safety-conscious work environment")
- **Finding and fixing problems** (The utility's corrective action program)

The review and assessment of these cross-cutting elements have an important role in the new program.

Overall Description

An overview of the reactor oversight process is shown below. For each cornerstone, NRC develops findings from inspections and licensees collect performance indicator data. Inspection findings are evaluated for safety significance using a significance determination process and performance indicator data is compared against prescribed risk-informed thresholds. The resulting information is then assessed and an appropriate NRC response is determined using the guidelines in an action matrix, which typically includes supplemental inspections for selected issues. Enforcement action is taken on significant inspection findings, as appropriate. NRC communicates the results of its performance assessment and its inspection plans and other planned actions in publically available correspondence, on its web site, and through public meetings with each licensee.



Measuring and Inspecting Nuclear Plant Performance

Nuclear plant performance will be measured by a combination of objective performance indicators and by the NRC inspection program. These will be closely focused on those plant activities having the greatest impact on safety and overall risk. In addition, the NRC will conduct both periodic and annual reviews of the effectiveness of each utility's programs to identify and correct problems.

Performance indicators use objective data to monitor performance within each of the "cornerstone" areas. The data which make up the performance indicators will be generated by the utilities and submitted to the NRC on a quarterly basis. Each performance indicator is measured against established thresholds which are related to their effect on safety. While performance indicators can provide insights into plant performance for selected areas, the NRC's inspection program provides a greater depth and breadth of information for consideration by the NRC in assessing plant performance.

The NRC will also monitor plant activities through its inspection program. The inspection program is designed to verify the accuracy of performance indicator information and to assess performance that is not directly measured by the performance indicator data.

Using Performance Indicators

Evaluation of Performance Indicator Data

The performance indicator data will be evaluated and integrated with findings of the NRC inspection program. Each of the performance indicators has criteria for measuring acceptable performance. (As in all industrial activities, nuclear power plants are not error-free or risk-free. Equipment problems and human errors will occur. Each performance indicator is designed to determine acceptable levels of operation within substantial safety margins.) These objective criteria are designed to reflect risk according to established safety margins, as indicated by a color coding system.

A "green" coding indicates performance within an expected performance level in which the related cornerstone objectives are met; "white" indicates performance outside an expected range of nominal utility performance but related cornerstone objectives are still being met; "yellow" indicates related cornerstone objectives are being met, but with a minimal reduction in safety margin; and "red" indicates a significant reduction in safety margin in the area measured by that performance indicator. The performance indicators will be reported to the NRC on a quarterly basis by each utility. Following compilation and review by the NRC staff, the performance indicators will be posted on the NRC's web site.

Performance Indicators by Cornerstone

Performance indicators are reported quarterly by operators of nuclear plants, reviewed by the NRC staff, and posted on the NRC's web site.

Safety Cornerstone	Performance Indicator
Initiating events	Unplanned reactor shutdowns (automatic and manual)
	Loss of normal reactor cooling system following unplanned shutdown
	Unplanned events that result in significant changes in reactor power
Mitigating Systems	Safety System not available • Specific Emergency Core Cooling Systems • Emergency Electric Power Systems
	Safety System Failures
Integrity of barriers to release of radioactivity	Fuel Cladding (measured by radioactivity in reactor cooling system)
	Reactor cooling system leak rate
Emergency Preparedness	Emergency response organization drill performance
	Readiness of emergency response organization
	Availability of notification system for area residents

Occupational Radiation Safety	Compliance with regulations for controlling access to radiation areas in plant
	Uncontrolled radiation exposures to workers greater than 10 percent of regulatory limit
Public Radiation Safety	Effluent releases requiring reporting under NRC regulations and license conditions
Physical Protection	Security system equipment availability
	Personnel screening program performance
	Employee fitness-for-duty program effectiveness

Inspection Programs

The revised oversight program continues to utilize a variety of NRC inspectors who monitor plant activities. The program includes baseline inspections common to all nuclear plants. The baseline inspection program, based on the "cornerstone" areas, focuses on activities and systems that are "risk significant," that is, those activities and systems that have a potential to trigger an accident, can mitigate the effects of an accident, or increase the consequences of a possible accident. The inspection program will also review the "cross-cutting issues" of human performance, the "safety-conscious work environment," and how the utilities find and fix problems. Inspections beyond the baseline will be performed at plants with performance below established thresholds, as assessed through information gained from performance indicators and NRC inspections. Additional inspections may also be performed in response to a specific event or problem which may arise at a plant.

The inspections will be performed by NRC resident inspectors stationed at each nuclear power plant and by inspectors based in one of the four NRC regional offices or in NRC headquarters in Rockville, Maryland. The regional offices are in King of Prussia, Pennsylvania; Atlanta, Georgia; Lisle, Illinois; and Arlington, Texas.

The new inspection program uses a "risk-informed" approach to select areas to inspect within each cornerstone. The inspection areas were chosen because of their importance from the point of view of potential risk, past operational experience, and regulatory requirements.

The baseline inspection program has three parts -- inspection of areas not covered by performance indicators or where a performance indicator does not fully cover the inspection area; inspections to verify the accuracy of a licensee's reports on performance indicators; and a thorough review of the utility's effectiveness in finding and resolving problems on its own.

Inspection reports will be issued for all inspections just as under the previous inspection program. The reports will be available to the public on the NRC's internet web site and from its Public Document Room at NRC headquarters.

Assessing Plant Performance

The inspection staff has developed a procedure, called the "Significance Determination Process," to help inspectors determine the safety significance of inspection findings. This process will be used for an initial screening review to identify those inspection findings that would not result in a significant increase in risk and thus need not be analyzed further (a "green" finding) . Remaining inspection findings -- which may have an effect on plant risk -- will then be subject to a more thorough risk assessment, using the next phase of the Significance Determination Process. This more detailed assessment may involve NRC risk experts from the appropriate regional office and further review by the utility's plant staff. The final outcome of the review -- evaluating whether the finding is green, white, yellow, or red -- will be used to determine what further NRC action may be called for.

Each calendar quarter, the resident inspectors and the inspection staff in the regional office will review the performance of all nuclear power plants in that region, as measured by the performance indicators and by inspection findings. Every six months, this review will be expanded to include planning of inspections for the following 12-month period.

Each year, the final quarterly review will involve a more detailed assessment of plant performance over the previous 12 months and preparation of a performance report, as well as the inspection plan for the following year. This review will include NRC headquarters staff members, the regional staff, and the resident inspectors.

These annual performance reports will be available to the public on the agency's web site, and the NRC staff will hold public meetings with utilities to discuss the previous year's performance at each plant.

In addition, NRC senior management will review the adequacy of agency actions for plants with significant performance problems. The managers will also take a wider view both of the overall industry performance and of the performance of the agency's regulatory programs. The performance of plants requiring heightened agency scrutiny will then be discussed during a public meeting with the NRC Commissioners at the agency's Rockville, Maryland, headquarters.

NRC Response Plan or "Action Matrix"	
Assessment of Plant Performance	NRC Response

(in order of increasing safety significance)	
I. All performance indicators and cornerstone inspection findings GREEN • Cornerstone objectives fully met.	• Routine inspector and staff interaction • Baseline inspection program • Annual assessment public meeting
II. No more than two WHITE inputs in different cornerstones • Cornerstone objectives fully met.	**Response at Regional level** • Staff to hold public meeting with utility management • Utility corrective action to address WHITE inputs • NRC inspection followup on WHITE inputs and corrective action
III. One degraded cornerstone (two WHITE inputs or one YELLOW input or three WHITE inputs in any strategic area) • Cornerstone objectives met with minimal reduction in safety margin	**Response at Regional level** • Senior regional management to hold public meeting with utility management • Utility to conduct self-assessment with NRC oversight • Additional inspections focused on cause of degraded performance
IV. Repetitive degraded cornerstone, multiple degraded cornerstones, or multiple YELLOW inputs, or one RED input • Cornerstone objectives met with longstanding issues or significant reduction in safety margin	**Response at Agency level** • Executive Director for Operations to hold public meeting with senior utility management • Utility develops performance improvement plan with NRC oversight • NRC team inspection focused on cause of degraded performance • Demand for Information, Confirmatory Action Letter, or Order
V. Unacceptable Performance • Unacceptable reduction in safety margin	**Response at Agency level** • Plant not permitted to operate • Commission meeting with senior utility management • Order to modify, suspend, or revoke license

NRC Response to Plant Performance

The quarterly reviews of plant performance, using both the performance indicators and inspection findings, will determine what additional action, if any, the NRC will take if there are signs of declining performance. This approach to enforcement is intended to be more predictable than previous practices by linking regulatory actions to performance criteria. The new process utilizes four levels of regulatory response with NRC regulatory review increasing as plant performance declines. The first two levels of heightened regulatory review are managed by the appropriate regional office. The next two levels call for an agency response, involving senior management attention from both headquarters and regional offices.

The oversight program retains the same tools used in the past for dealing with declining plant performance and violations. These tools, however, are used in a more predictable manner that is commensurate with the decreased safety performance. In the past, the NRC tended to use fines as a prime indicator of agency concern and as a motivator to affect licensee corrective actions. Under the new approach, there is a system of specified agency actions if performance declines. Fines will generally be reserved for such things as discriminating against workers raising safety concerns, or willful misreporting of required information.

The NRC's actions for performance below the "green" level may include meetings with the utility, additional inspections, and required reviews and response by the utility. Further declines in performance would warrant stronger action by the NRC, including a civil order or even the suspension of the utility's operating license.

Violations of NRC Requirements

Each violation of NRC requirements found during NRC inspections will be evaluated to determine its effect on plant safety and risk. If the violation is of very low safety significance, it will be discussed in the inspection report with no formal enforcement action. The utility is expected to deal with the violation through its corrective action program, correcting the violation and taking steps to prevent a recurrence. The issue may also be reviewed during future NRC inspections.

If the NRC risk evaluation finds that the violation has higher safety significance, a Notice of Violation will be issued. A Notice of Violation may also be issued if the utility fails to correct a violation of low safety significance in a reasonable period of time or if a violation is found to be willful.

The Notice of Violation requires the utility to respond formally to the NRC with its actions to correct the violation and what steps it will take to prevent the violation from occurring again. The agency will then review the utility's actions in a later inspection.

Normally, these violations will not be the subject of a fine. However, there may be violations that warrant a fine because of their unusual significance. These violations are likely to be uncommon. Possible examples include exceeding a safety limit specified in a reactor license or the inadvertent startup of a reactor.

In addition, some violations will call for the traditional enforcement approach, including the possible issuance of fines. Examples include:

• Discrimination against workers for raising safety issues or other willful violations.

• Actions that may adversely affect the NRC's ability to monitor utility activities, including failure to report required information, failure to obtain NRC approval for plant changes, failure to maintain accurate records, or failure to provide the NRC with complete and accurate information.

• Incidents with actual safety consequences, including radiation exposures above NRC limits, releases of radioactive material above NRC limits, or failure to notify government agencies when emergency response is required.

Communications/Making Information Available to the Public

The revised oversight process will provide more information on plant performance than in the past, and the information will be available on a more frequent basis. This information will be placed on the NRC's internet web site as well as in its Public Document Room at NRC headquarters.

A utility will submit to the NRC the quarterly performance indicator data for each nuclear power plant it operates. The NRC staff will review the data for completeness and accuracy. The staff will also evaluate inspection findings for that quarter to determine their safety significance. This review uses the agency's "Significance Determination Process," which is keyed to how plant safety systems and procedures contribute to the risk of a potential accident.

The performance indicators and the assessment of inspection findings will be placed on the NRC web site using the color notation of their significance -- green, white, yellow, or red. The statistics and inspection findings which underlie the color notation will also be posted on the web site.

The revised oversight program is intended to fulfill the following four goals established by the Commission:

1. To maintain safety by establishing a regulatory oversight framework that provides assurance that plants continue to be operated safely by plant operators. Maintaining safety is the NRC's overarching mission.

2. To enhance public confidence in the NRC's regulatory program by increasing the predictability, consistency, objectivity and transparency of the oversight process so that all parties will be well served by the changes taking place.

3. To improve the effectiveness, efficiency, and realism of the oversight process by focusing both agency resources and utility resources on those issues with the most safety-significance.

4. To reduce unnecessary regulatory burden as the process becomes more efficient and effective.

How This New Oversight Process Differs from the Previous Approach

The previous oversight process evolved over a period of time when the nuclear power industry was less mature and there was much less operational experience on which to base rules and regulations. Very conservative judgments governed the rules and regulations. Significant plant operating events occurred with some frequency, therefore the oversight process tended to be reactive and prescriptive, closely observing plant performance for adherence to the regulations and responding to operational problems as they occurred.

But we now have the benefit of four decades of operational experience and, generally speaking, steadily improving plant performance, particularly over the last decade or so. The new program focuses more of the agency's resources on the relatively small number of plants which evidence performance problems. The baseline inspection program is considered the minimum inspection effort needed to assure that plants meet the "safety cornerstone" objectives. The baseline inspection program is performed at all reactor sites by NRC resident inspectors and inspectors from the regional offices.

Plants which do not meet the "safety cornerstone" objectives, measured by performance indicators and inspection findings, will receive increased inspection, focusing on areas of declining performance. There will also be inspections beyond the baseline program, even at plants performing well, if there are operational problems or events the NRC believes require greater scrutiny. Generic problems, affecting some or all plants, may also require additional inspections.

The previous oversight program relied more heavily on fines when violations occurred, while the new program will make broader use of other enforcement tools such as orders and other formal regulatory actions. When fines were imposed previously, they were often issued long after the violations occurred and their impact was substantially less than the cost of repairs or the costs associated with a shutdown to correct the violations. The new process is intended to be more effective in correcting performance or equipment problems because the agency's response will be both more timely and more predictable.

The new assessment program is substantially different from the previous process. It makes greater use of objective performance indicators. Together, the indicators and inspection findings provide the information needed to support reviews of plant performance, to be conducted on a quarterly basis, with the results posted on the NRC's internet site.

The new assessment process also features expanded reviews on a semi-annual basis to include inspection planning and a performance report, all of which will also be posted on the NRC's web site.

The performance assessment process previously involved three processes:

- **Plant Performance Review** - Conducted every six months to assess events, inspection findings, and other data. This review was done to plan future inspections and to identify those plants with declining performance that required further NRC action.

- **Senior management meetings** - Twice a year, NRC senior managers reviewed information assessing plant performance to discuss what regulatory action was needed at plants with declining performance. The managers designated those plants warranting heightened NRC monitoring as being on a "watch list." These "watch list" plants were then discussed at a public meeting with the Commission.

- **Systematic Assessment of Licensee Performance** (SALP) - Every 12 to 24 months, the NRC staff performed a separate review of the performance of each plant, preparing a Systematic Assessment of Licensee Performance report. This report included a numerical rating of the plant in four categories -- plant operations, maintenance, engineering, and plant support -- as well as a narrative discussion of performance in each area.

Glossary

Baseline Inspection Program - The normal inspection program performed at all nuclear power plants. The program will focus on plant activities that are not adequately measured by performance indicators, on the corrective action program, and on verifying the accuracy of the performance indicators.

Corrective Action Program - The system by which a utility finds and fixes problems at the nuclear plant. It includes a process for evaluating the safety significance of the problems, setting priorities in correcting the problems, and tracking them until they have been corrected.

Cross-cutting Area - Nuclear plant activity that affects most or all safety cornerstones. These include the plant's corrective action program, human performance, and "safety-conscious work environment."

Inspection Reports - Reports are issued periodically to document inspection findings. These may cover a specific time period for the baseline inspection or a particular event or problem examined in a reactive inspection. All inspection reports are public documents and, when issued, are posted to the NRC's internet web site.

Performance Indicator - Objective data which records performance in a specific cornerstone of safety at a nuclear power plant.

Reactive Inspection - An inspection to examine the circumstances surrounding an operational problem or event occurring at a nuclear plant.

Regulatory Conference - A meeting between the NRC staff and a utility to discuss potential safety issues or to discuss a change in performance as indicated by a declining performance indicator or inspection finding. These meetings are open to public observation unless they cover security issues, NRC investigation findings, or similar sensitive topics.

Resident Inspector - An NRC inspector assigned to a nuclear plant on a full-time basis. Each site has at least two resident inspectors.

Risk-informed - Incorporating an assessment of safety significance or relative risk in NRC regulatory actions

Cornerstone of Safety - Nuclear plant activities that are essential for the safe operation of the facility. These cornerstones are grouped under the categories of reactor safety, radiation safety, and safeguards.

Safety Conscious Work Environment - A working environment in which employees are encouraged to report safety concerns without fear of criticism or retaliation from their supervisors because they raised the issue.

Significance Determination Process - The process used by the NRC staff to evaluate inspection findings to determine their safety significance. This involves assessing how the inspection findings affect the risk of a nuclear plant accident, either as a cause of the accident or the ability of plant safety systems or personnel to respond to the accident.

Updated July 14, 2000

ATTACHMENT 3

January 9, 2001

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

EA-00-208

Garry L. Randolph, Senior Vice
President and Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, Missouri 65251

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR THREE WHITE FINDINGS AND NOTICE OF VIOLATION (NRC INSPECTION REPORT 50-483/00-17, CALLAWAY PLANT)

Dear Mr. Randolph:

The purpose of this letter is to provide you with the final results of our significance determination of the preliminary White findings identified in the subject inspection report. The inspection findings were assessed using the significance determination process and were preliminarily characterized as three White findings (i.e., issues with low to moderate increased importance to safety, which may require additional NRC inspections).

The findings involved performance deficiencies in your ALARA (As Low As is Reasonably Achievable) planning and controls program. We emphasize that, although there were no exposures in excess of regulatory limits, the performance deficiencies resulted in unnecessary doses to workers during Refueling Outage 10. As documented in the subject inspection report, these deficiencies involved: 1) planning and conducting maintenance activities in the vicinity of the reactor coolant system (RCS), during a time period soon after shutdown, when area dose rates were temporarily elevated by a chemical cleaning process designed to remove radioactive particulate from RCS internal surfaces, without commensurate compensatory measures; 2) planning and conducting maintenance activities in the vicinity of the steam generators before the steam generator bowl drains were flushed, resulting in higher than normal area dose rates without commensurate compensatory measures; 3) conducting maintenance activities on the reactor coolant pumps and steam generators without the steam generator secondary sides filled with water, resulting in higher than normal area dose rates without commensurate compensatory measures; 4) conducting maintenance activities without sufficient mock-up training to familiarize contract workers with plant equipment, use of tools, and techniques to effectively reduce the dose that they would receive; and 5) performing maintenance activities with ineffective communications between radiation protection personnel and the primary contractor, which resulted in additional worker exposure due to ineffective planning and sequencing of work activities. Your staff originally estimated that plant workers would receive exposures totaling 165 person-rem during Refueling Outage 10. The actual value was 305 person-rem. Your staff discussed a number of factors to explain the differences between the actual and estimated values. Notwithstanding, the NRC concluded that a significant portion of this increase was the result of poor ALARA practices.

At your request, a regulatory conference was held on November 9, 2000, to discuss your views on this issue. During the meeting, your staff described your assessment of the significance of the findings, corrective actions, and the root cause evaluations for the issues. You provided supplemental information in a letter dated November 16, 2000, in which you took issue with the NRC's determination of the process control level at which a work activity should be defined as a "job." The job classification is used for the purpose of calculating the amount of excess dose accumulated and consequently characterizing the significance of a finding in accordance with the Occupational Radiation Safety Significance Determination Process (SDP). Based on your interpretation of Callaway Plant procedures, you asserted that the Work Authorizing Document (WAD) is the appropriate process control level that should be used to classify a particular activity as a job for ALARA purposes, and that, utilizing this approach, the findings appeared to constitute one White finding, rather than the three White findings which were identified by the NRC in the subject inspection report.

Notwithstanding that assertion, after considering the information developed during the inspection, the additional information you provided at the regulatory conference, and the information provided in your November 16, 2000, letter, the NRC has concluded that the inspection findings are appropriately characterized as three White findings. We recognize that the term "job" is not formally defined by the SDP and its supporting guidance. However, as discussed in the November 9, 2000, regulatory conference, the term "jobs" in the Occupational Radiation Safety SDP clearly corresponds to those work activities for which distinct ALARA planning and controls are implemented. From our review of your procedure PDP-ZZ-00003, "Work Document Planning," Rev. 28, and your conduct of in-progress job and post-job reviews required by procedure HTP-ZZ-01102, "Pre-Job ALARA Planning and Briefing," Rev. 14, we conclude that your ALARA planning and controls were primarily implemented at the Radiation Work Permit (RWP) level rather than at the WAD level for the work activities in question. For ALARA purposes, Callaway Plant procedures allow multiple WADs to be grouped and controlled under one RWP. Consequently, the bases for the three White findings described in the inspection report remain valid.

The first White finding involved scaffolding activities (RWP-50903). We noted that for scaffolding activities, dose projections were made for the RWP, in-progress reviews were conducted for the RWP, and post-job reviews were conducted for the RWP. None of these activities occurred for the associated scaffold permits or the associated WAD. Since this RWP accrued more than 25 person-rem and exceeded its dose projection by greater than 50 percent, it constituted a single White finding.

The second White finding involved steam generator eddy current/robotic plugging/stabilizing/ electrosleeving activities (RWP-53323). Although dose projections were made for the associated WADs, there were no work process information sheets completed for each WAD. Similarly, an in-progress job review was done for the RWP, not the individual WADs, and post-job reviews were performed for the RWP, and not the individual WADs. Again, since this RWP accrued more than 25 person-rem and exceeded its dose projection by greater than 50 percent, it constituted a second White finding.

The third White finding occurred because there were four jobs with actual doses greater than 5 person-rem and exceeded their dose projections by more than 50 percent. These jobs included steam generator manway covers and inserts removal and installation (RWP 99-53321), health physics support for primary and secondary steam generator activities (RWP 99-53324), foreign object search and retrieval (RWP 99-53022), and reactor coolant pump seal removal and replacement (RWP 99-52520). ALARA planning and controls were instituted for these four RWPs, and not their associated WADs.

We acknowledge that the performance associated with these findings occurred before April 1, 2000, the implementation date of the revised reactor oversight program (ROP). However, we are assessing these findings in a manner consistent with the ROP initial year implementation guidance which directs that findings identified in inspection reports completed after April 1, 2000, will be assessed under the ROP regardless of when the performance deficiency occurred.

You have 10 business days from the date of this letter to appeal the staff's determination of significance for the identified White findings. Such appeals will be considered to have merit if they meet the criteria given in NRC Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 0609.03.

The NRC has also determined that these demonstrated performance deficiencies constitute a violation of 10 CFR 20.1101(b). Specifically, you did not use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses ALARA. The violation is cited in the attached Notice of Violation (Notice), and the circumstances surrounding the violation are summarized in this letter and described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice of Violation is considered an escalated enforcement action because it is associated with White findings.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

Because plant performance for these findings has been determined to be in the degraded cornerstone column of the operating reactor assessment Action Matrix, we will notify you, by separate correspondence, of our determination of the appropriate NRC response.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

Union Electric Company

/RA/

Ellis W. Merschoff
Regional Administrator

Docket No.: 50-483
License No.: NPF-30

Enclosure: Notice of Violation

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NOTICE OF VIOLATION

Union Electric Company
Callaway Plant

Docket No. 50-483
License No. NPF-30
EA-00-208

During an NRC inspection conducted on August 7-11, 2000, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> 10 CFR 20.1101(b) requires that the licensee use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).
>
> Contrary to the above, during Refueling Outage 10, conducted between October and November 1999, the licensee did not use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses ALARA. Specifically, although the original dose estimate for Refueling Outage 10 indicated that plant workers would receive exposures totaling 165 person-rem, the actual dose received was 305 person-rem and a significant portion of this increase was attributable to poor ALARA work practices. For example:

a. the licensee planned and conducted maintenance activities in the vicinity of the reactor coolant system (RCS), during a time period soon after shutdown, when area dose rates were temporarily elevated by a chemical cleaning process designed to remove radioactive particulate from RCS internal surfaces, without commensurate compensatory measures, resulting in doses that were not ALARA.

b. the licensee planned and conducted maintenance activities in the vicinity of the steam generators before the steam generator bowl drains were flushed, resulting in higher than normal area dose rates without commensurate compensatory measures, resulting in doses that were not ALARA.

c. the licensee conducted maintenance activities on the reactor coolant pumps and steam generators without the steam generator secondary sides filled with water, resulting in higher than normal area dose rates without commensurate compensatory measures, resulting in doses that were not ALARA.

d. the licensee conducted maintenance activities without sufficient mock-up training to familiarize contract workers with plant equipment, use of tools, and techniques to effectively reduce the dose that they would receive.

e. the licensee performed maintenance activities with ineffective communications between radiation protection personnel and the primary contractor, which resulted in additional worker exposure due to ineffective planning and sequencing of work activities.

Union Electric Company

This violation is associated with three White SDP findings.
Pursuant to the provisions of 10 CFR 2.201, Union Electric Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region IV, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available to the Public, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you are required to post this Notice within two working days.

Dated this 9th day of January 2001

Union Electric Company

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Callaway Plant

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February 7, 2001

U. S. Nuclear Regulatory Commission
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ULNRC-4378

Gentlemen:

APPEAL OF FINAL SIGNIFICANCE DETERMINATION
INSPECTION REPORT NO. 50-483/00-017
CALLAWAY PLANT
UNION ELECTRIC CO.

Ref: 1) ULNRC-4298, dated August 21, 2000
2) EA-00-208, dated October 4, 2000
3) ULNRC-4343, dated November 16, 2000
4) EA-00-208, dated January 9, 2001

Union Electric appeals the NRC Staff's final significance determination as set forth in the letter of January 9, 2001, from Ellis W. Merschoff, Regional Administrator, Region IV, to Garry L. Randolph, Senior Vice President and Chief Nuclear Officer, Union Electric, for the three findings identified under NRC Inspection Report 50-00483/00-17. This appeal should be given consideration by an appeal panel as the significance determination was both inconsistent with the applicable Significance Determination Process (SDP) guidance and lacked justification (appeal category 3.b, NRC Inspection Manual Chapter 0609, Attachment 3).

On January 23, 2001, Mr. William D. Johnson, Chief, Division of Reactor Projects, Branch B, advised Union Electric that the deadline for submitting this appeal had been extended to February 7, 2001.

Pursuant to the requirements of 10 CFR 2.201, Union Electric will submit, under separate cover, a written response to the Notice of Violation (NOV) accompanying the January 9, 2001, letter.

To preface the ensuing discussion, with noted exceptions, the Reactor Oversight Process (ROP) has exhibited marked improvement over the former inspection and enforcement process with regard to objectivity, scrutability, regulatory focus on risk significance, and the reduction of unnecessary regulatory burden. Throughout the implementation and transitional period into the ROP, the Staff has endeavored to maintain strict adherence to the program as designed. These efforts to preserve the integrity of the process should yield more meaningful observations regarding the usability, effectiveness and consistency of the ROP in light of the upcoming comment period for the first year of implementation. In addition, strict adherence to the ROP guidelines has generally provided for a more predictable and consistent characterization of inspection findings within an inspected area and, to a limited degree, from area to area across the spectrum of the inspection program.

In approving implementation of the revised ROP and termination of the previous assessment process, Systematic Assessment of Licensee Performance, the Commission noted that this action "will inevitably reveal issues that were not exposed in the pilot program. The [NRC] staff should anticipate that adjustments – perhaps significant adjustments – will be necessary as the program unfolds. As a result, there should be a continuing open dialogue with NRC licensees, other stakeholders, and staff, as issues are encountered." This appeal is a formal part of that dialogue anticipated by the Commission.

The issues discussed in detail in this appeal have a direct and adverse impact on the integrity of the ALARA portion of the ROP and, subsequently, the goals of objectivity, scrutability, focus on risk significance and the reduction of unnecessary regulatory burden. Union Electric believes that the SDP for ALARA is fatally flawed and should be suspended until it can be revised to be consistent with existing regulatory requirements and the goals of the ROP. The SDP for ALARA improperly assigns "low to moderate safety significance" to collective occupational doses that have no safety significance. It is subjective, inscrutable, less predictable, does not focus on risk significance and creates a new regulatory burden. The SDP creates a new and different duty on licensees for their radiation exposure control programs. As implemented by the NRC Staff, it is new or different from a previously applicable staff position without the systematic and documented analysis required by 10CFR 50.109.

Even if the NRC were to conclude that the new SDP for ALARA is not fatally flawed and should be enforced, it was not properly applied at Callaway. In particular, the ALARA planning function that resulted in the noted deficiencies occurred prior to the adoption of the new SDP for ALARA, resulting in an *ex post facto* application of a new requirement. This violated any notion of due process because the ALARA planning for

Refueling Outage 10 was performed under standards different from those contained in the new SDP for ALARA. As a minimum, the NRC should have appropriately analyzed "jobs" to reflect Union Electric's intent that the Work Authorizing Document is the lowest level of ALARA planning. Even if the SDP for ALARA were to be applied retroactively, an appropriate evaluation of the five examples cited in the NOV should result in findings of "no color."

Union Electric understands that the SDP for ALARA resulted from a well-intentioned attempt to establish a metric for inspection of the ALARA programs at nuclear plants. The new Regulatory Oversight Process has made many needed improvements to the inspection and enforcement at nuclear plants and Union Electric strongly supports the effort. Union Electric's experience with the SDP for ALARA, however, suggests that it is inconsistent with the risk-informed basis of the Regulatory Oversight Process and counter-productive to the intent of ALARA.

As a matter of clarification it should be noted the 165 person-rem goal discussed in the Final Significance Determination letter was not the refuel dose projection. As noted at the Regulatory Conference the Dose projection was 210 person-rem based on planned Work Authorizing Documents for Refueling Outage 10.

The details of this appeal are included in Attachment 1. None of the material in this appeal is considered proprietary by Union Electric.

If you have any questions regarding this response, or if additional information is required, please contact me.

Very truly yours,

G. L. Randolph

GLR/JVL/MAR/mib

Attachment: 1) Appeal of Final Significance Determination

cc: Mr. Ellis W. Merschoff
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Superintendent, Licensing
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ATTACHMENT 1

APPEAL OF FINAL SIGNIFICANCE DETERMINATION ASSOCIATED WITH ALARA NOTICE OF VIOLATION

(NRC INSPECTION REPORT 50-483/00-17, CALLAWAY PLANT)

This enclosure describes in detail the bases for this appeal to the NRC Staff's determination of significance for the identified three White findings set forth in a letter dated January 9, 2001 from Ellis W. Merschoff, Regional Administrator, Region IV, to Garry L. Randolph, Senior Vice President and Chief Nuclear Officer, Union Electric Company. Union Electric will respond separately, as required, to the Notice of Violation (NOV) accompanying the January 9 letter.

INTRODUCTION AND SUMMARY

In general, Union Electric does not disagree that there were areas requiring improvement in its performance of ALARA controls during Callaway Refueling Outage 10 in October 1999. All of the information set forth in the NOV was self-identified by Union Electric prior to the NRC inspection during August 2000. Before Refueling Outage 10 was completed, Union Electric requested an Institute of Nuclear Power Operations (INPO) Assist Visit to evaluate the performance of its ALARA program, which was conducted in January 2000. Furthermore, Union Electric conducted a thorough self-assessment with industry peer evaluators in June 2000. A formal root cause evaluation was completed in November 2000. Performance deficiencies and corrective actions were entered into the Callaway Corrective Action Program. Lessons learned are being incorporated into the planning for Callaway Refueling Outage 11. Union Electric considers escalated enforcement by the NRC inappropriate for having an ALARA program that can find and initiate issues in execution of work.

Union Electric disagrees with the NRC Staff's application of the new Significance Determination Process (SDP) as it has been applied to ALARA planning and controls, both in general and specifically at Callaway. The cornerstone activity under consideration is Occupational Radiation Safety, which is to ensure adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation.[1] No worker exceeded a regulatory limit nor a Callaway administrative limit for dose during Refueling Outage 10. The cited examples of ALARA work practices in the NOV were not precursors to exceeding individual exposure limits. Consequently, there were no health or safety impacts of the identified deficiencies relating to ALARA controls.[2] Yet, a White

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1 NRC Inspection Manual Chapter 0609, Appendix C

2 Union Electric previously articulated its position on the lack of safety significance of the ALARA inspection findings in a letter on August 21, 2000 from R. D. Affolter (Manager,

Footnote continued on next page

finding indicates issues that are of "'low to moderate safety significance."[3] The issues at Callaway do not meet that definition under any circumstance. The areas for improvement in ALARA controls that Union Electric identified at Callaway were not desirable and Union Electric has taken aggressive action to correct them. Nevertheless, they simply do not represent any safety significance.

Our research has not identified an enforcement action under previous inspection and enforcement policies relating to inspection of an ALARA program at a commercial nuclear facility. This is not surprising because all nuclear plant licensees have an ALARA program[4] and conduct their operations and maintenance activities in a manner generally consistent with that program. The SDP for ALARA actually creates a new regulatory requirement – dose estimates for radiation work permits must be accurate. This interpretation of 10 C.F.R. § 20.1101(b) is without doubt "new or different from a previously applicable staff position."[5] Commission precedent and due process in implementing administrative changes dictate that changes in a Commission policy may not create a new regulatory requirement.

Union Electric believes that the issues addressed in this appeal have a direct and adverse impact on the integrity of the ALARA portion of the new Reactor Oversight Process (ROP) and its goals of objectivity, scrutability, focus on risk significance and the reduction of unnecessary regulatory burden. Careful consideration of these issues calls into question the appropriateness and justification of the SDP for ALARA. Union Electric concludes that the SDP for ALARA is fatally flawed and should be suspended until it can be revised to be consistent with existing regulatory requirements and the goals of the ROP.

Even if the NRC concludes that the new SDP for ALARA is not fatally flawed and should be enforced; for the reasons discussed in the second section of this letter, it was not applied appropriately at Callaway in any event. In particular, the ALARA planning function that resulted in the noted deficiencies occurred prior to the adoption of the new SDP for ALARA, resulting in an *ex post facto* application of a new requirement. This violates any notion of due process, as the ALARA planning function at Callaway being evaluated was performed before the new ALARA SDP established the parameters to be monitored.

Footnote continued from previous page

Callaway Plant) (ULNRC-4298) to the NRC. The NRC has not yet responded to Union Electric's argument.

3 NRC Inspection Manual Chapter 0305, paragraph 04.05

4 As currently required by 10 C.F.R. § 20.1101(b), and, prior to 1991, suggested by 10 C.F.R. § 20.1(c).

5 *See,* 10 C.F.R. § 50.109(a)(1) (which defines backfitting as "modification of or addition to ... procedures or organizations required to design construct or operate a facility; any of which may result from ... the imposition of a regulatory staff position interpreting the Commission rules that is either new or different from a previously applicable staff position ....")

Union Electric understands that the SDP for ALARA resulted from a well-intentioned attempt to establish a metric for inspection of the ALARA programs at nuclear plants. The new Regulatory Oversight Process has made many needed improvements to the inspection and enforcement at nuclear plants and Union Electric strongly supports the effort. Union Electric's experience with the SDP for ALARA, however, suggests that it is inconsistent with the risk-informed basis of the Regulatory Oversight Process and counter-productive to the intent of ALARA.

BACKGROUND ON REACTOR OVERSIGHT PROCESS (ROP) AND ALARA SIGNIFICANCE DETERMINATION PROCESS (SDP)

To understand why Union Electric considers the SDP for ALARA inconsistent with the ROP, it is valuable to describe the origins and goals of the new ROP. On March 9, 1998, the NRC issued SECY-98-045, "Status of the Integrated Review of the NRC Assessment Process for Operating Commercial Nuclear Reactors," which recommended a new integrated assessment process. This process would be based on inspection findings characterized by safety significance that would be scored into performance template areas. Assessment would be based on comparing the totaled scores against a threshold and NRC action taken based on a decision model. In parallel, the nuclear industry, led and coordinated by the Nuclear Energy Institute (NEI), developed an approach that was fundamentally and philosophically different. Industry proposed an assessment process that used high-level, objective indicators in performance areas like maintaining the integrity of barriers to radioactivity release. Each indicator would have thresholds set to form a utility response band, a regulator response band and a band of unacceptable performance.[6]

Based on comments from the Commission, from a public hearing and from a Senate hearing, the NRC staff set out to develop a single set of recommendations for making improvements to the regulatory oversight processes.[7]

On January 8, 1999, the NRC issued SECY-99-007, "Recommendations for Reactor Oversight Process Improvements", which recommended a framework for regulatory oversight with seven cornerstones of safety. Licensee performance that met the objectives and key attributes of each of these cornerstones would provide reasonable assurance that public health and safety were met.[8] Within each cornerstone, there would be performance indicators and results of inspections that will have risk-informed thresholds. Crossing these thresholds would be based on safety significance and would prompt a need for some NRC interaction.[9]

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6 SECY-99-007

7 SECY-99-007

8 SECY-00-0049

9 SECY-99-007

On March 22, 1999, the NRC issued SECY-99-007A, "Recommendations for Reactor Oversight Process Improvements (Follow-up to SECY-99-007)," which presented a plan to conduct a pilot of the new assessment process at eight sites.[10] Part of the pilot would test a Significance Determination Process (SDP) which would screen issues identified during inspections to elevate potentially risk-significant issues, to screen out issues of minimal or no risk significance and to trigger more detailed analysis when warranted.[11] At that time, the ALARA SDP had not been drafted, but would be completed in time to support the pilot.[12]

A six month pilot was conducted at eight sites between May 30, 1999, and November 27, 1999.[13] On February 24, 2000, the NRC issued SECY-00-0049, "Results of the Revised Reactor Oversight Process Pilot Program," which recommended implementing the revised ROP, but indicated that further experience with the process is needed. The NRC approved implementation of the revised ROP and termination of the previous assessment process, Systematic Assessment of Licensee Performance (SALP), but noted that this action "will inevitably reveal issues that were not exposed in the pilot program. The [NRC] staff should anticipate that adjustments – perhaps significant adjustments – will be necessary as the program unfolds. As a result, there should be a continuing open dialogue with NRC licensees, other stakeholders, and staff, as issues are encountered."[14]

Versions of the Occupational Exposure SDP were issued on August 10, 1999, and November 12, 1999.[15] During the pilot program, the SDP, including the ALARA SDP, was not exercised across the full range of potential inspection findings.[16] On April 21, 2000, the SDP was issued in a finalized form, including the ALARA SDP.[17] Under the current SDP, observations noted during an inspection are evaluated by an initial assessment to determine if it

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10 SECY-99-007A

11 SECY-99-007A

12 SECY-99-007A

13 SECY-00-0049

14 SRM of May 17, 2000, "Staff Requirements – SECY-00-0049 – Results of the Revised Reactor Oversight Process Pilot Program (Part 2)"

15 NEI comments on SDP, Attachment 4. These comments noted a need to address clarity of job dose screening criteria and extent of aggregation of multiple green findings to find significance.

16 SECY-00-0049, Attachment 6

17 NRC Inspection Manual Change 00-007, issued April 21, 2000.

is a "finding".[18] Findings are then evaluated by the SDP and Enforcement Review Panel (SERP)[19] in accordance with procedures described in cornerstone specific appendices.

In general, observations are classified as minor issues that can be discussed with the licensee, but do not merit documentation based on inspector judgment.[20] If an issue is not considered minor, the issue is analyzed to determine whether it affects a cornerstone. For ALARA issues to be considered to impact a cornerstone, they must satisfy all of three criteria related to the job and the licensee's average collective dose.[21] If the issue does not affect a cornerstone (for ALARA issues, not meet any one of the three criteria), then it should normally not be documented, unless evaluated and determined to have extenuating circumstances.[22]

For those ALARA issues that are determined to affect a cornerstone, they are sorted into the Green, White, Yellow or Red significance band.[23] A Green finding is one associated with a job where the actual collective dose is less than 25 person-rem, unless there have been three or more such occurrences in the last 18 months. If there have been at least three occurrences, the finding is White. The finding can also be White, if associated with a job where the actual collective dose is greater than 25 person-rem, unless the plant's current three-year rolling

18 NRC Inspection Manual Chapter 0609, paragraph 08.02

19 NRC Inspection Manual Chapter 0609, Attachment 1 is intended to describe the SERP, but has not yet been finalized.

20 NRC Inspection Manual Chapter 0609, Attachment 2. This attachment lists seven questions that can be used by an inspector, in general, to determine if an issue is minor, such as "Could the issue be viewed as a precursor to a significant event?" The attachment also references another appendix for guidance as it "is the most recent information and best examples of what constitutes minor issues."

21 NRC Inspection Manual Chapter 0609, Attachment 2 calls these criteria "Group 2 Questions". For ALARA issues, the Group 2 Questions are: "(a) Does the actual job dose exceed the projected dose by >50%, AND (b) [does] the 3 year rolling average collective dose exceed 135 person-rem/unit for a PWR or 240 person-rem/unit for a BWR, AND (c) is the actual job dose > 5 person-rem?"

22 NRC Inspection Manual Chapter 0609, Attachment 2 calls the criteria to determine extenuating circumstances "Group 3 Questions" and lists six, including "Is the finding a violation?"

23 NRC Inspection Manual Chapter 0609, Appendix C. NRC Inspection Manual Chapter 0305, paragraph 04.05 defines the risk significance associated with each color, ranging from Green (not desirable, very low) through White (low to moderate) and Yellow (substantial) to Red (unacceptable loss of safety margin, high).

average exceeds 340 person-rem (for PWRs) or 600 person-rem (for BWRs). In which case the finding is Yellow. There are no criteria for Red ALARA findings.[24]

I. THE ALARA SDP IS FATALLY FLAWED AND SHOULD BE SUSPENDED

Audits, inspections and assist visits prior and subsequent to Callaway Refueling Outage 10 identify that Union Electric has a strong ALARA program overall, which is capable of finding and correcting weaknesses in ALARA controls. All of the information set forth in the Notice of Violation was self-identified.[25] Also, Union Electric noted the need to upgrade its ALARA performance independent of NRC's inspection findings. Although undesirable, none of the identified areas for improvement in ALARA work practices indicate a potential to exceed individual worker regulatory or licensee administrative exposure limits. Importantly, no worker exceeded individual regulatory or Callaway administrative exposure limits during Refueling Outage 10.[26] As such, there is no safety significance to the ALARA practices cited. Higher than desired aggregate occupational exposure should be avoided, but, on its own, does not indicate a problem of any safety significance, much less "low to moderate safety significance" as represented by a White finding.

The SDP for ALARA fails to meet the ROP goals of objectivity, scrutability, focus on risk significance and the reduction of unnecessary regulatory burden. The SDP for ALARA is subjective, inscrutable, less predictable, does not focus on risk significance, and creates a new burden.

The SDP for ALARA effectively creates a "new and different duty" for licensees relating to their radiation exposure control programs. The new requirement for accuracy in dose estimating is "new and different from a previously applicable staff position" without the "systematic and documented analysis" required by 10 C.F.R. § 50.109. In other words, the SDP for ALARA results in an impermissible backfit.

Consistent with the action taken by the NRC for the Fire Protection SDP, the NRC should suspend use of the ALARA SDP until these fatal flaws can be corrected.

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24 NRC Inspection Manual Chapter 0609, Appendix C

25 Compare Union Electric's report, *Refuel 10 ALARA Outage Report (October 2, 1999 to November 5, 1999)* issued in June 2000 with NRC Inspection Report 50-483/00-17 of October 4, 2000, and NRC Notice of Violation EA-00-208 of January 9, 2001.

26 Callaway Plant's dose data for 1999, including Refueling Outage 10, shows that no individual exceeded 40 percent of the individual dose limits, greater than 95 percent of individuals receiving measurable dose accumulated less than 20 percent of the limits, and greater than 80 percent of individuals receiving measurable dose accumulated less than 10 percent of the individual limits. No worker exceeded a regulatory limit nor a Callaway administrative limit for dose during Refueling Outage 10. *See also,* Occupational Radiation Exposures at NRC Licensed Facilities (1999), NUREG-0713, Table 4.6 and Appendix B, page B-2.

A. The Callaway ALARA Program: Strengths and Areas for Improvement

NRC Inspections and peer evaluations of the Callaway ALARA program consistently found a strong, effectively implemented program prior to the August 2000 NRC Inspection. A March 2000 NRC Inspection looked, in part, at Union Electric ALARA performance during Refueling Outage 10 and made findings consistent with a May 1998 NRC inspection that concluded Union Electric had a very good ALARA program effectively implemented. An August 2000 NRC Inspection looked at Union Electric ALARA performance during Refueling Outage 10 and came to a dramatically different conclusion. The Callaway ALARA program did not change during the period between NRC inspections. Rather, the new NRC metric that was applied for the first time evaluated something entirely different than the inspections had addressed in the past.

One of the key features of a strong ALARA program is the ability to find areas requiring improvement and take corrective action. Compliance with the ALARA requirement is judged on whether the licensee has incorporated measures to track, and, if necessary, reduce exposures.[27] One reason for setting aggressive exposure goals prior to a job is to provide a screen to identify jobs, which warrant additional scrutiny for possible corrective actions. At Callaway, the ALARA program was working as designed; aggressive dose projections led to identifying many jobs from Refueling Outage 10 where improvements could be made.[28] These areas for potential improvement were investigated and corrective actions initiated, prior to the August 2000 NRC inspection. The five cited examples of Refueling Outage 10 ALARA work practices identified in the NOV had all been investigated by Union Electric prior to the NRC inspection and corrective action had been initiated where appropriate. The NRC inspection report and NOV do not take issue with any of the corrective actions Union Electric has taken or planned.

During Refueling Outage 10, Union Electric recognized that the execution of work did not maintain aggregate occupational exposure as low as desired and requested assistance from INPO to focus on the ALARA program. The INPO Assist Visit was conducted in January 2000 and a list of actions for consideration was issued by INPO in February 2000. The INPO Assist Visit team proposed actions in seven focus areas, including source term reduction, scheduling and planning, and ALARA process reviews.[29] These focus area actions cover the five examples

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27 *See, Standards for Protection Against Radiation – Final Rule*, 56 Fed. Reg. 23360 at 23367 (1991), "Compliance with this requirement will be judged on whether the licensee has incorporated measures to track and, if necessary, to reduce exposures and not whether exposures and doses represent an absolute minimum or whether the licensee has used all possible methods to reduce exposures."

28 *See*, Union Electric's report, *Refuel 10 ALARA Outage Report (October 2, 1999 to November 5, 1999)*, issued in June 2000.

29 INPO (Steven L. Driscoll, Manger, Radiation Protection Programs) letter to Union Electric of February 4, 2000.

of Refueling Outage 10 ALARA work practices noted in the NRC NOV of January 9, 2001, almost a year before the NRC issued its NOV. Union Electric developed plans to implement actions in the seven focus areas, which were available for NRC review. The NRC did not raise any issues with the planned actions during the August 2000 inspection.

Union Electric conducted a peer review of the ALARA program at Callaway utilizing personnel from Callaway and three other nuclear power plants for a week in June 2000 and established detailed action plans. The self-assessment focused on incorporation of ALARA into the planning of work, but also touched on daily dose budgeting and incorporation of ALARA into supervisory pre-job briefs. The review generated eighteen Suggestion/Occurrence/Solution (SOS) action documents to address needed improvements.[30] This review was completed almost two months before the NRC ALARA inspection. The NRC did not raise any issues with the planned actions during the August 2000 inspection.

A formal root cause evaluation was completed in November 2000, prior to NRC initiating enforcement action. Performance improvements and corrective actions were entered into the Callaway Corrective Action Program. Lessons learned are being incorporated into the planning for Callaway Refueling Outage 11.[31]

The NRC conducted an inspection of the radiation protection activities at Callaway in March 2000 and, with regards to ALARA, noted only that exposure trends were increasing, attributable to increased outage work scope and increased source term from an axial offset anomaly. The NRC inspection reviewed Callaway dose totals and averages for the three previous years. In addition to increased refueling outage work scope, the increasing trend in doses was attributed to a higher source term, which was exacerbated by a reactor fuel condition known as an axial offset anomaly.[32] Union Electric actions in response to this anomaly had been previously evaluated by the NRC as conservative and in accordance with regulatory requirements in December 1997.[33] These March 2000 NRC findings are consistent with the May 1998 NRC inspection, which concluded that Union Electric had a very good ALARA program, effectively implemented.

The Callaway ALARA program and performance during Refueling Outage 10 did not change during the period between NRC inspections in March and August 2000. Rather, the new NRC metric that was applied for the first time evaluated something entirely different than the inspections had addressed in the past. It is the ALARA SDP that is fatally flawed and, therefore,

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30 Union Electric Report SA00-HP-001, ALARA Work Planning/Support and Radiation Worker Knowledge Self-Assessment dated June 16, 2000.

31 *See, e.g.*, AmerenUE presentation on collective radiation dose at Regulatory Conference of November 9, 2000, Slide 25.

32 NRC Inspection Report 50-483/00-07 of March 28, 2000.

33 NRC Inspection Report 50-483/97-19 of December 18, 1997.

does not provide justification for assigning significance of three white findings to the cited ALARA work practices self-identified by Union Electric.

B. The SDP for ALARA improperly assigns "low to moderate safety significance" to collective occupational doses with no safety significance

This new NRC metric which assigns significance to the failure to achieve all ALARA collective exposure goals, is not consistent with NRC regulatory requirements and past NRC policy. Occupational exposure to radiation has health and safety significance, but the appropriate measure of that significance is dose to the individual worker. To reflect this, the NRC has established limits for the exposure to an individual worker, but no limits for the aggregate exposure to all workers. Although undesirable, none of the Refueling Outage 10 cited ALARA practices indicate a potential to exceed individual worker regulatory or licensee administrative exposure limits. As such, there is no health or safety significance to the ALARA practices noted. Higher than desired or planned aggregate occupational exposure should be avoided, but it does not indicate a problem of any health or safety significance, and certainly not one of "low to moderate safety significance," unless there is an indication of a potential for individual workers to exceed limits.

When the ALARA rule was revised in 1991, the NRC focus remained on individual exposure within limits. The NRC agreed there would be advantages to establishing a floor, below which efforts to further reduce collective exposure would be left to licensee ALARA programs without NRC oversight.[34] At that time, the NRC was evaluating whether the floor should be established for collective exposures at either 100 or 1000 person-rem.[35] The new ALARA SDP codes an ALARA finding as White or "low to moderate safety significance" if the associated job is 25 person-rem and exceeds its estimate by 50%, or if at least three jobs of 5 person-rem exceed their estimates by 50%. The licensee must provide dose estimates within 50% for all but two jobs over 5 person-rem and any job over 25 person-rem. In other words, a White finding can be assigned for excess collective exposures of 9 person-rem on one job or a sum of three jobs having excess collective exposures of 5 person-rem total. The NRC has not justified why collective exposures around 10 person-rem now warrant enhanced NRC enforcement attention when just 10 years ago collective exposures at least an order of magnitude higher were considered below regulatory concern as long as individual exposure limits were met.

The worker exposures at Callaway during Refueling Outage 10 were properly documented and controlled. Although Union Electric agrees that the aggregate exposures were higher than desired, none of the ALARA practices identified at Callaway indicate the potential for any worker to exceed occupational exposure limits. The relatively small excess exposure that any individual worker received during Refueling Outage 10 is not of any recognized health risk.

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34 *See, Standards for Protection Against Radiation – Final Rule*, 56 Fed. Reg. 23360 at 23366 (1991).

35 *See, Policy Statement on Below Regulatory Concern*, 55 Fed. Reg. 27522 (1990).

NRC regulations establish limits for individual occupational exposure and for individual members of the public, not for aggregate population (collective) dose. The ALARA concept is an important part of an adequate radiation protection program. Due to the practice of maintaining radiation exposures ALARA, the average worker's dose is well below limits. This is consistent with the concept of the ALARA regulation as intended to be an operating principle.[36] The NRC regulations require that licensees use procedures and engineering controls, to the extent practical, to maintain occupational and public exposures ALARA.[37] These regulations require licensees to have and follow a process to minimize exposure, without specifying a particular outcome. Collective dose measurement and assessment is an inexact indicator of the success of a licensee's process unless it is evaluated with judgment and experience.[38]

Operating a nuclear power plant safely will require some occupational exposure; the amount depends on balancing the risks of exposure against the need to do maintenance and modifications to ensure safe operation and to operate the plant economically. These decisions involve many non-quantifiable or non-fungible factors, including nuclear safety risk, ALARA goals, and operating or maintenance costs. For example, the ALARA regulations do not establish an equivalency between person-rem and dollars. The NRC has avoided adopting any requirement for a numerical cost-benefit analysis to demonstrate ALARA, as many ALARA procedures reflect sound operating practice which are not suitable for numerical analysis. Furthermore, the cost to conduct the numerical analysis may exceed, in some cases, the value of the dose reduction.[39] Any such equivalency could only be a rough guideline to inform decisions and would need to be applied with judgment. The ALARA regulations mandate that the judgments be informed by sound radiation protection principles, but do not mandate a specific result.

The flaw in using a mechanistic screen of average collective dose to determine safety significance is illustrated by the experience at Callaway. The process concludes that the facts at Callaway result in a violation of "low to moderate safety significance" because the rolling average is over the PWR threshold, but the same facts would be of no safety significance if

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36 *See, Standards for Protection Against Radiation – Final Rule*, 56 Fed. Reg. 23360 at 23366 (1991), "This shift is to emphasize that the ALARA concept is intended to be an operating principle rather than an absolute minimization of exposures."

37 10 C.F.R. § 20.1101(b).

38 *See, e.g.*, NRC Inspection Manual, Inspection Procedure 71121, Attachment 2, paragraph 02.06, which states, "The significance of ALARA findings will often depend on reasonably accurate exposure estimates. Reasonable implies that they be based on good assumptions and correct calculations with some flexibility given with regard to expected variability due to the limits of forecasting."

39 *See, Standards for Protection Against Radiation – Final Rule*, 56 Fed. Reg. 23360 at 23367 (1991).

Callaway were a BWR. Callaway currently has a three-year rolling average of annual exposure of 178 person-rem, which exceeds the average for PWR plants (and Union Electric is committed to lowering the average.) In fact, lowering the rolling average was the basis for the 165 person-rem goal established for Refueling Outage 10.[40] There can be no health or safety significance to the magnitude of Callaway's rolling average, however, as it is less than the 240 person-rem average for BWR plants. The higher total exposures at Callaway compared to other PWRs are traceable, in significant part, to a higher source term due to axial offset anomaly and steam generator work as noted by the NRC in March 2000.[41] Union Electric's decisions to address these potential safety concerns and incur higher aggregate exposures reflect a properly informed judgment. It is a reflection of the success of the ALARA program at Callaway that these exposure totals are not higher in the face of higher maintenance workloads and plant design issues, related to steam generator work and axial offset anomaly, compared to other PWRs. As an illustration, during Refueling Outage 10, Union Electric shifted steam generator maintenance strategy from electrosleeving to plugging for the last two of the four steam generators. This change reduced collective dose for Refueling Outage 10. (Cost and schedule savings were also factors in the decision along with ALARA concerns.)[42]

Consistent with sound science and past NRC policy, collective dose should not have independent significance under the Occupational Radiation Safety Cornerstone. ALARA findings should be considered significant when they are a precursor indication of potential for exceeding occupational limits. The Performance Indicator for the cornerstone counts events on control of high or very high radiation levels or of unintended individual exposures a significant fraction of limits. This indicator properly focuses on the potential to exceed individual occupational exposure. ALARA inspection findings can also play a role as predictors of potential issues of safety significance when evaluated with judgment. The purely mechanistic approach of the ALARA SDP metric does not accomplish that aim.

C. The SDP for ALARA is subjective, inscrutable, less predictable, does not focus on risk significance and creates new burden

The NRC Strategic Plan establishes that one of the performance goals is zero significant radiation exposures from civilian nuclear reactors.[43] Consistent with this goal, the objective of the Occupational Radiation Safety Cornerstone is to ensure worker health and safety from exposure to radiation and this objective is obtained by maintaining worker doses within the NRC

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40 *See, e.g.*, AmerenUE presentation on collective radiation dose at Regulatory Conference of November 9, 2000, Slide 5.

41 NRC Inspection Report 50-483/00-07 of March 28, 2000.

42 The collective dose for electrosleeving two of four steam generators was 24.3 person-rem, providing an indication of the likely collective exposure reduction for canceling the work on the last two steam generators.

43 SECY-99-007, page 7.

individual exposure limits and ALARA.[44] As such the objective is to minimize individual exposures, consistent with NRC regulations that establish individual limits.[45] The potential that individual exposures were not minimized to the extent practical is one explanation for higher than expected aggregate exposures on a job. Other factors, not under the licensee's control, are also likely explanations, such as higher dose rates or job scope growth. In most cases, the exposure increase will be due to a mixture of these factors. In evaluating the significance of an ALARA finding as an indicator of degraded cornerstone performance, the SDP metric must differentiate between factors under the licensee's control and those that are not or the goal of revised Reactor Oversight Process to improve focus on aspects of licensee performance will not be met.

An ALARA finding that is caused by licensee performance and is related to individual exposure issues could provide advance warning of a potential issue of safety significance. However, a mechanistic application of aggregate exposure totals, does not differentiate between issues that impact individual exposures and those that do not. Simply put, not every issue that causes a job to exceed its exposure estimate by 50% is a failure of an ALARA program and should be addressed by NRC enforcement attention. Some issues are the result of aging plants requiring more maintenance or plant specific design problems. If ALARA issues that cannot be effectively addressed are given the same significance as those that can be, the effectiveness of the Occupational Radiation Safety Cornerstone as an indicator of where NRC attention should be focused will not achieve its purpose.

In addition to differentiating significance to those ALARA issues that are under licensee control, ALARA issues that are not related to an increased risk of individual exposure exceeding limits need to be differentiated from those that do. This significance cannot be determined from a mechanistic evaluation of total job exposure and a given percent over projection. For example, poor practices such as failure to provide appropriate temporary shielding could be a potential contributor to an individual's exposure exceeding limits, if the work was near sources of very high dose rates. Without allowing application of judgment in determining the significance of an ALARA finding, there will be no differentiation between ALARA findings of safety significance and those with no safety significance.

ALARA observations that are classified as minor issues based on inspector judgment do not merit documentation.[46] Since not all ALARA issues that warrant documentation have a direct or an indirect impact on safety, this presents a conundrum for the inspector, as he cannot document any ALARA issues unless they have impact on safety. The basis for concluding the issues in the August 2000 NRC Inspection Report are of safety significance has not been

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44. NRC Inspection Manual Chapter 0609, Appendix C.

45. Plain language definition of the Cornerstone, <www.nrc.gov/NRR/OVERSIGHT/ROP/description.html>

46. NRC Inspection Manual Chapter 0609, Attachment 2.

adequately documented or justified.[47] Contrary to NRC guidance, neither the inspection report nor the Final Significance Determination letter provides justification as to why the identified findings are not minor,[48] nor do they address the alternative perspectives provided by Union Electric under ULNRC-4298, dated August 21, 2000. Although, it is unclear how the findings were characterized for evaluation against these questions, the NRC stated during the exit meeting conducted August 11, 2000, that the basis for justifying documentation was that the observations have an actual or credible impact on safety. Given this assumption, Union Electric is concerned that NRC Inspectors will be compelled to conclude that any excess dose over the SDP metric has some impact on safety in order to justify documenting their ALARA observations. This will have an adverse impact on the consistency and validity of inspection reports. The threshold for safety associated with occupational radiation exposure would be established by where each inspector designates an ALARA issue as more than minor. Historically, this has not been the case, as any dose within occupational limits was considered of no health significance. This new interpretation would represent a significant shift in the Staff's position on the significance of collective dose.

The revised ROP established a graded matrix of risk-informed thresholds to assure that safety margins are being maintained and that sufficient time exists for both the NRC and licensees to address noted performance deficiencies before there was an undue risk to health or safety.[49] The safety significance of an ALARA occupational exposure finding is the potential health risk to the worker from increased exposure. Unless the finding indicates a potential for exceeding individual limits or the ALARA process is significantly malfunctioning, there has been no significant degradation in safety margins. NRC inspection guidance recognizes that the ALARA rule does not require every ALARA effort to demonstrate optimized exposure performance.[50] Failing to minimize exposure to the extent practical in all cases is undesirable,

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47 Union Electric previously articulated its position on the lack of safety significance of the ALARA inspection findings in a letter on August 21, 2000 from R. D. Affolter (Manager, Callaway Plant) (ULNRC-4298) to the NRC. The NRC has not yet responded to Union Electric's argument.

48 NRC Inspection Manual Chapter 0610* requires an inspection report to have a certain level of detail, which includes in part "...a significance evaluation paragraph that describes the logic for entering the SDP. That is, it answers the pertinent group 1,2 or 3 'thresholds for documentation' questions". This requirement is also referenced in NRC Inspection Manual Chapter 0609, attachment 3, which states "The basis should allow a knowledgeable reader to duplicate the logic that resulted in the staff's significance determination. In cases where the staff is aware of a licensee's alternative perspectives, the staff should give its justification for not accepting the licensee perspectives in the basis discussion."

49 SECY 00-0049, page 16.

50 NRC Inspection Manual, Inspection Procedure 71121, Attachment 2, paragraph 03.05. *See also*, *Minor Corrections, Clarifying Changes, and a Minor Policy Change*, 63 Fed. Reg. 39477 (1998), which changed the work "practicable" to "practical" in the ALARA regulation (10 C.F.R. § 20.1101(b)) to remove the basis for an incorrect perception that NRC is requiring

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but it does not indicate a degraded ALARA process if the utility learns from the event and applies appropriate corrective actions.

Besides not meeting the goals of the revised ROP to be more focused on safety, the SDP metric for ALARA has the unintended consequence of making NRC enforcement action more subjective and less predictable. The goals of the revised ROP included making the oversight process more objective, more scrutable and more focused on safety.[51] The SDP for ALARA in practice enhances focus on inspector subjectivity. First, an ALARA issue is subject to screening. During screening, inspector's judgment is allowed only in deciding whether or not an issue is minor or not. Once the inspector has decided an issue warrants being documented, the mechanistic SDP criteria take over. To avoid this conundrum, inspectors may have varying thresholds in deciding which ALARA observations should be discussed with the licensee and which should be documented and essentially escalated by the SDP for ALARA.[52] Which finding goes in which category could be subject to wide and unpredictable variation between inspectors.

The importance of whether an issue is documented is magnified by the SDP process. Once documented, a mechanistic screening process takes over that is likely to conclude there is "low to moderate safety significance." By their very nature ALARA findings will involve more than one job. An inspector who notes a poor ALARA work practice on only one job is unlikely to consider the issue worth documenting as the NRC inspection manual recognizes that ALARA does not require every job to be optimized. Likely, most of the ALARA issues evaluated by the ALARA SDP will be associated with multiple jobs. As the NRC Inspection Manual specifies that the inspector focus on jobs meeting the criteria for potentially White significance,[53] any ALARA issue documented is likely to be considered of "low to moderate safety significance."

Several unusual results can come from the SDP for ALARA. For example, if a PWR has a rolling three-year average less than 135 person-rem (240 for a BWR), all findings are assigned, "no color", even the most severe ALARA violation. If one programmatic issue, as is the case at Callaway, is associated with more than one job, multiple colors can be assigned to that one

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licensees to use any conceivable dose averting technique, even if the technique is unproven or impractical.

51 SECY 99-007, page 6.

52 *See, e.g., NRC's Color-Coded PI System Still Murky, Says State Regulator,* Inside NRC, page 15 (January 29, 2001).

53 NRC Inspection Manual, Inspection Procedure 71121, Attachment 2, paragraph 02.05.

event. This result is contrary to the stated intent from the ROP Pilot Program that each single event be given only one color significance, even if manifesting itself in multiple categories.[54]

The SDP does not provide a definition of job,[55] identify what "job associated with an ALARA finding" means or identify how many jobs should be used to find the significance of any one ALARA finding. The licensee cannot anticipate how the inspector will combine or divide up jobs and obtain a consistent and predictable significance between sites or between inspections at the same site. Whether a degraded cornerstone is found depends on what the inspector considers worth documenting and how many jobs he looks at. The unintended consequence is to shift focus to NRC inspection methods and away from licensee performance. For example, Union Electric procedures expect ALARA planning and controls to be implemented at the Work Authorizing Document (WAD) level.[56] The NRC considers that the

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[54] SECY 00-0049, attachment 9 (*Assessment Process*) pages 1 and 2, which identifies that one event, a reactor trip, should be evaluated as one White finding, even though it is counted under two categories within a cornerstone.

[55] *See*, NRC letter to Union Electric of January 9, 2001, which states, "We recognize that the term 'job' is not formally defined by the SDP and it's supporting guidance. ... [T]he term 'jobs' in the SDP clearly corresponds to those work activities for which distinct ALARA planning and controls are implemented."

[56] In support of Union Electric's position, their work instructions and notes included in the WAD are to consider ALARA concerns, such as:

- Component removal to a low dose area.
- Prefab work outside the RCA.
- Part or component replacement, rather than repair, in areas with significant radiation levels.
- Mockup training.
- Component and/or WAD history review to determine lessons learned that may be included in the work instructions, or need further follow-up by the shop or planner prior to working the job.
- Job site walk downs to identify problems or work site interferences.
- Include information in work instructions for "Troubleshoot" WRs that minimize worker time in Radiation Areas.
- Items such as prints, location drawings, vendor manual excerpts, information regarding work history, and suspected problems.

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definition of job is at the Radiation Work Permit (RWP), inconsistent with the intent of Union Electric's procedures for ALARA planning and control at Callaway.[57] Union Electric does not disagree with the NRC observation, and considers planning at the RWP level a part of its practice. Shifting more planning to the WAD level is one of the corrective actions Union Electric desires to implement. Union Electric is concerned that the further improvement by the nuclear industry toward achieving ALARA will be hampered by the subjective and unpredictable interpretations adopted during NRC inspections.

D. The SDP for ALARA creates a new and different duty on licensees for their radiation exposure control programs.

Administratively, the SDP is nothing more than guidelines for NRC Inspectors. In practice, the SDP creates a new duty on licensees, as they strive to comply with ALARA regulations. The ALARA NOV at Callaway is coded with a safety significance of three White findings, since there were several RWPs that exceeded the SDP mechanistic guidelines for total job exposure and percentage over estimates. Focusing enhanced enforcement attention on this issue can only dilute regulatory attention from issues with safety significance and engender baseless worker concerns over the health effects of small individual exposures. The significance of an ALARA finding should be determined considering corrective actions taken by the licensee. Further, the focus should be on licensee performance. The SDP should consider whether factors beyond the licensee's control, like plant design problems and unique maintenance needs, contribute to the above average performance. The SDP, by providing for enhanced enforcement without considering mitigating factors, like licensee corrective actions and without focusing on licensee performance, brings a new and arbitrarily higher significance to ALARA deficiencies. This radical shift in enforcement priorities for ALARA is not justified by the previous 30-year success of the ALARA regulations.

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- Work locations listed on the WAD should be correct. A map or drawing and ALARA or Surrogate Tour photos should also be included in the WAD, when available.

In addition to the instructions provided to the discipline planners, the Rad/Chem Planner and station ALARA personnel also review each Work Authorizing Document prior to placing it into the Health Physics computer system. This is a second level review of the Work Authorizing Document to consider any additional ALARA planning and controls that may be desired. Consequently, a "job" at Callaway Plant is the Work Authorizing Document as all ALARA planning starts at this level in the organization.

57 NRC letter to Union Electric of January 9, 2001, which states, "from our review of your procedure PDP-ZZ-00003, 'Work Document Planning,' Rev. 28, and your conduct of in-progress job and post-job reviews required by procedure HTP-ZZ-01102, 'Pre-Job ALARA Planning and Briefing,' Rev. 14, we conclude that your ALARA planning and controls were primarily implemented at the Radiation Work Permit (RWP) level rather than at the WAD level for the work activities in question".

Assigning a "low to moderate safety significance" to the ALARA NOVs is not justified if the mitigating factors involved are considered. For example, as discussed above, the ALARA issues documented in the NRC inspection report had been found previously by Union Electric at Callaway. The NRC did not identify any deficiencies in Union Electric's planned corrective actions, which indicates that licensee action to improve ALARA performance without extra attention from Region IV should be adequate.

Previously, no ALARA NOVs were issued, but the enforcement policy stated that the failure to maintain and implement a program was a level IV violation (lowest of four levels). The new ALARA SDP metric would require one aspect of ALARA process, exposure estimating, to be executed almost perfectly to avoid White findings. Putting ALARA budget adherence on the same safety footing as plant safety work is an unprecedented involvement by the NRC in licensee operating discretion. Barrier integrity improvements could be delayed, to reduce plant rolling average and avoid the chance of White ALARA findings. Any PWR with a rolling three-year average over 340 person-rem or BWR over 600 would be encouraged to avoid scheduling 25 person-rem jobs to avoid a chance of an ALARA finding assigned Yellow significance. The significance associated with the Callaway ALARA NOV should reflect the importance of preserving the spirit, not just the letter, of maintaining licensee discretion in plant operations where appropriate.

E. **As implemented by the NRC Staff, the SDP for ALARA is "new or different from a previously applicable staff position" without the "systematic and documented analysis" required by 10 C.F.R. § 50.109.**

ALARA has worked well over the past 30 years by relying on licensee-managed procedures. Converting ALARA enforcement from ensuring there is an effective process "based on sound radiation protection principles" (as required by 10 C.F.R. § 20.1101) to requiring quantified, outcome-specific, aggregate exposures is a "new or different" position from "a previously applicable staff position."[58] This new position has not been justified under the required analyses for backfits.

The ALARA program at Callaway had been consistently evaluated as a strong, effectively implemented program. After the implementation of the SDP for ALARA in April 2000, the August 2000 NRC Inspection found three White findings of "low to moderate safety significance," which Union Electric considers unjustified. The examples of ALARA work practices warranting improvement were not of safety significance per accepted science or past NRC policy. It is the subjective and unpredictable criteria of the SDP for ALARA that lead to this surprising result. Licensees now have a new and different duty to provide precise dose estimates or face escalated enforcement.

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58 10 CFR § 50.109

The consequence of assigning a new and arbitrarily higher significance to ALARA issues is a *de facto* backfit resulting solely from the implementation of the ALARA SDP.[59] Licensees are now required to maintain dose projections in a way that is almost completely specified by the NRC regarding the activities covered, the method and justifications for revisions, and the tracking and documentation required, in order to have an "accurate" assessment from the NRC. The licensee's discretion is restricted to managing the ALARA program within the constraints specified or implied by the SDP. The result being that ALARA planning and controls implementation methodologies have been specified by the NRC (through the SDP) without regard for how the licensee wishes to implement the program. Regarding the current situation at Callaway, the NRC has effectively relieved the Union Electric staff of its ability to manage the ALARA program in the most effective and efficient manner suited to its organizational structure and philosophies. Specifically, the net effect at Callaway is that the site-wide organization has been relieved of its responsibility for managing ALARA and that responsibility has been shifted squarely to the formal ALARA review process, which is solely the responsibility of the health physics staff.

The NRC has not conducted a systematic and documented analysis to justify the imposition of the new duty on licensees, effectively restricting the amount of work the licensee can perform at any PWR to that which results in no more than 135 person-rem, on average, per year.[60] Any work scope that results in a higher average exposure and results in excess collective exposure on any exposure on the order of magnitude of 10 person-rem, exposes the licensee to liability for a degraded occupational safety cornerstone. There are potential negative implications for safety-related maintenance and upgrade work if licensees defer those activities to fall within the ALARA SDP metric. It is this type of undisciplined change with potential far-reaching consequences that the NRC was trying to avoid in revising the backfit rule.[61]

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59 *See, Revision of Backfitting Process for Power Reactors*, 50 Fed. Reg. 38097 at 38101 (1985), which states, "there is no practical difference between a backfit that is imposed pursuant to a rule or a staff position."

60 To avoid the potential for ALARA issues to impact a safety cornerstone, PWR licensees must not exceed 135 person-rem under NRC Inspection Manual Chapter 0609, Attachment 2. BWR licensees are allowed a higher average. There is no principled reason to expose BWR workers to collective doses higher than PWRs. The SDP for ALARA establishes a rough distinction between two classes of reactors, but allows no fine distinction for site specific differences.

61 *See, Revision of Backfitting Process for Power Reactors*, 50 Fed. Reg. 38097 at 38101 (1985), which states, "changes in ... staff positions for procedures and organizations should also be analyzed before implementation to determine, inter alia, the safety significance of any such proposed change."

Conclusion: The new SDP for ALARA is fatally flawed and should be suspended

The new ALARA SDP as implemented is fatally flawed as it assigns an inappropriate level of significance to imperfections in estimating and executing work. Union Electric executed about 3000 jobs during Refueling Outage 10. The NRC recognizes that the ALARA regulations do not mandate that collective exposure for all work be the absolute minimum. Progress in reducing each worker's exposure is made possible by developing improvement actions by investigating jobs where the exposure returns exceed the original estimates. The NRC is penalizing Union Electric for having an aggressive ALARA program that finds and implements improvements in ALARA work practices.

No worker exceeded a regulatory limit nor a Callaway administrative limit for dose during Refueling Outage 10. The cited examples of ALARA work practices were not such as to be precursors to exceeding individual exposure limits. Consequently, there was no health or safety impacts of the identified deficiencies relating to ALARA controls. The areas for improvement in ALARA controls that Union Electric identified at Callaway were not desirable and Union Electric has taken aggressive action to correct them. Nevertheless, they simply do not represent any safety significance.

The SDP for ALARA actually creates a new regulatory requirement – dose estimates for radiation work permits must be accurate. There are potential negative implications for safety-related maintenance from establishing an effective ceiling on work of 135 person-rem per year. The SDP for ALARA is an impermissible backfit.

Consistent with the action for the Fire Protection SDP,[62] the NRC should suspend use of the ALARA SDP until these fatal flaws can be corrected.

II. THE SDP FOR ALARA WAS INCORRECTLY AND RETROSPECTIVELY APPLIED AT CALLAWAY SO THAT THE NRC STAFF'S SIGNIFICANCE DETERMINATION IS INCONSISTENT WITH THE APPLICABLE SDP GUIDANCE.

Even if the NRC were to disagree with the foregoing discussion regarding the fatal flaws in the ALARA SDP, Union Electric disagrees with its application at Callaway. It was inappropriately and unfairly applied retroactively and it was applied incorrectly in a manner inconsistent with the applicable SDP guidance. The NRC found only one violation at the Callaway plant, that the licensee did not use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses as low as reasonably achievable during Refueling Outage 10. Since Refueling Outage 10 was planned and conducted a year before the ALARA SDP was issued, it violates any notion of due process

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62 *See*, SECY-00-0049, which discusses the need to upgrade several SDPs prior to initial implementation, including the Fire Protection SDP which had already been issued. *See generally,* history of implementing the Physical Protection SDP (which based on Union Electric's knowledge and belief was also suspended).

and fairness to apply the ALARA SDP to a finding from that outage; therefore, the violation should be considered a "no color" finding. In the alternative, even if the ALARA SDP is applied retroactively, it should be applied in its current form only once to the one violation, and find the significance as one "no color" finding. This result is achieved by assigning significance of an appropriate aggregate exposure for Refueling Outage 10 or by evaluating the unnecessary exposure associated with each of the five examples of Refueling Outage 10 ALARA work practices listed in the violation. Applying the ALARA SDP to six Radiation Work Permits (RWPs) from Refueling Outage 10 incorrectly implies that any RWP involving over 5 person-rem that exceeds its original estimate by 50% is a "finding," contrary to the definition of "finding" in the NRC Inspection Manual. In addition, applying the definition of job at the RWP rather than Work Authorizing Document (WAD) level is inconsistent with the intent of Union Electric's procedures for ALARA planning and control at Callaway.

A. Retroactive application of the new ALARA SDP to Refueling Outage 10 is inappropriate and unfair

Applying the ALARA SDP metric for planning performance to Refueling Outage 10 at all is an impermissible retroactive application of an enforcement policy, even if the ALARA SDP were not suspended as we recommend. The SDP represents an impermissible new or different NRC staff position on ALARA,[63] as previously there was no regulatory significance to job exposure estimates that were unrealistically low. Past NRC inspection policy sought to ensure that exposure estimates did not become inflated.[64] The ALARA SDP reflects a new policy of requiring exposure estimates to be accurate. This policy was applied to Callaway Refueling Outage 10, even though the SDP had not been issued prior to the outage and Union Electric had no notice of this change in policy at the time of its ALARA planning.[65]

A previous NRC inspection of Callaway and the prior history of NRC enforcement established ALARA violations as low safety significance. The NRC inspection at Callaway in

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63 *See,* 10 C.F.R. § 50.109(a)(3) (which limits backfitting to only when, with regard to the overall protection of public health and safety or the common defense and security, there is either a finding of necessity or documented analysis showing a substantial increase in protection justifying the costs to implement.)

64 *See, e.g.*, NRC Inspection Manual, Inspection Procedure 71121, Attachment 2, paragraph 02.06.a and b, which emphasizes that job exposure estimates should be verified as reasonable by comparison with past site-specific experience, industry data, and actual exposure results.

65 The NRC letter of January 9, 2001, acknowledges that the performance associated with these findings occurred prior to the implementation date of the revised ROP, but that the ALARA SDP is to be applied to inspection reports completed after April 1, 2000, under the guidance for the ROP initial year implementation. This guidance was issued in February 2000, six months after Refueling Outage 10 was completed. At the time Union Electric planned and executed the outage, it had no notice that its collective exposure estimates and returns would be subject to scrutiny under a different standard.

March 2000 with regards to ALARA noted only that exposure trends were increasing, attributable to increased outage work scope and to a higher source term, which was exacerbated by a reactor fuel condition known as an axial offset anomaly.[66]

Under the NRC Enforcement Policy, ALARA violations are Level IV, while White findings indicate a significance equivalent to Level III. White findings under the revised ROP require a response comparable to that required for Level III violations under the enforcement policy. If Union Electric had known the revised importance of using precisely accurate estimates, it would have taken actions to update the estimates during Refueling Outage 10. Instead, Union Electric made a decision to maintain the original estimates in the face of mounting evidence during the outage that the estimates were too low, in part to magnify the problem areas of ALARA adherence in work execution at Callaway.[67]

The estimates used by Union Electric for Refueling Outage 10 were appropriate for ALARA purposes. They were aggressive estimates, and their effectiveness as tools to spark investigations to discover less than optimum implementation of ALARA work practices was enhanced by being somewhat unrealistic, as they were not adjusted for higher than expected dose rates. Since the estimates were aggressive, more jobs were investigated for potential improvements to ALARA practices. The effectiveness of this process is illustrated by the number of issues noted during the INPO and peer reviews conducted following Refueling Outage 10 and the number of improvement actions developed for implementation.

In 1991, when the ALARA regulation was revised from a hortatory suggestion to a mandatory requirement, the NRC noted that maintaining the emphasis on ALARA as an operating principle should reduce potential problems in retrospective evaluation of licensee performance.[68] The new ALARA SDP reverses course and no longer maintains focus on whether there is a program to conduct ALARA reviews and efforts made to achieve ALARA. Instead, the new ALARA SDP focuses on whether specific exposure returns are achieved compared to pre-job estimates. It was this sort of retroactive enforcement of a mandate to ensure the absolute minimization of exposures that the NRC was trying to avoid in its adoption of 10 C.F.R. Section 20.1101 as a final rule.[69] Consistent with Commission precedent and due process

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66 NRC Inspection Report 50-483/00-07 of March 28, 2000.

67 Union Electric letter from R. D. Affolter (Vice President, Nuclear) (ULNRC-4343) to the NRC of November 16, 2000, Attachment 1, page 8.

68 *Standards for Protection Against Radiation – Final Rule*, 56 Fed. Reg. 23360 at 23367 (1991).

69 *See, Standards for Protection Against Radiation – Final Rule*, 56 Fed. Reg. 23360 at 23366 (1991), which states in response to comments about problems in the retrospective evaluation of licensee performance by NRC inspectors, "The emphasis on ALARA actions has been revised from detailed requirements to document all ALARA actions to a requirement to have a radiation protection program that includes measures to keep doses and intakes 'as low as is reasonably achievable.' This shift is to emphasize that the ALARA concept is intended to be

Footnote continued on next page

in implementing administrative changes,[70] even if the ALARA SDP is to be applied, it should not be applied to work planned and executed before it was issued.

If the ALARA SDP were not applied to the Callaway ALARA NOV, the NOV would certainly be assigned one finding of "no color". This designation appears appropriate for the first ALARA NOV ever issued. Attachment 2 to NRC Inspection Manual 0609 states that an issue will be documented as a "no color" finding in those cases involving extenuating circumstances.

B. Even if the SDP for ALARA were to be applied retroactively to Refueling Outage 10 at Callaway, significance should be based on Union Electric's performance on the entire outage.

If the ALARA SDP were to be applied retroactively to the Callaway ALARA NOV, it should be applied programmatically and conclude there was one "no color" finding. One example of an aggregate measure is that Refueling Outage 10 had a 210 person-rem budget that was exceeded by 47 person-rem, which is the most exposure attributable to ALARA work practices and is not more than 25% over budget. In accordance with the SDP for ALARA, the result is a "no color" finding, as the applicable group 2 question from Attachment 2 to NRC Inspection Manual 0609 would be answered that the actual job dose does not exceed projected dose by greater than 50%. The NRC states that during Refueling Outage 10, the licensee did not achieve ALARA, " as [Union Electric] originally estimated that plant workers would receive exposures totaling of 165 person-rem during Refueling Outage 10. The actual value received was 305 person-rem ... a significant portion of the increase was attributable to poor ALARA work practices."[71] As the violation is for the entire outage, the significance of the violation

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Footnote continued from previous page

an operating principle rather than an absolute minimization of exposures."

70 Retroactivity is defined as the taking away or impairing of vested rights acquired under existing law, or creating a new obligation, imposing a new duty or attaching a new disability in respect to transactions or consideration already past. (*Association of Accredited Cosmetology Schools v. Alexander*, 979 F.2d 859, 864 (D.C. Cir. 1992)). In this matter, the ALARA SDP creates a new duty of accuracy in dose estimating and assigns a new disability, enhanced enforcement, to degraded ALARA work practices. "Retroactivity is not favored in the law. Thus congressional enactments and administrative rules will not be construed to have retroactive effect unless their language requires this result." (*Bowen v. Georgetown University Hospital*, 488 U.S.204, 208 (1988)). "Retroactive impositions of civil liability are conceptually of a piece with *ex post facto* criminal laws," and the court finds the reordering of affairs inherently repugnant (*Ralis v. RFE/RL, Inc.*, 770 F.2d 1121, 1126 (D.C. Cir. 1985)). *But see*, other cases have held that retroactivity is to be assumed unless otherwise stated when lower court decisions are reviewed on appeal, *see, e.g., Bradley v. Richmond School Board*, 416 U.S. 696, 711 (1974) which states that "a court is to apply the law at the time it renders its decision."

71 NRC letter to Union Electric of January 9, 2001.

should be based on an aggregate measure of Union Electric's performance during the outage, not by selectively analyzing performance on a few jobs from the 3000 completed during the outage.

The NRC Significance Determination incorrectly identifies that Union Electric originally estimated that plant workers would receive exposures totaling 165 person-rem during Refueling Outage 10. In fact, 165 person-rem was the management goal established over a year prior to the start of the exposure estimating process. The budget established at the completion of the dose estimating process at the start of the Refueling Outage was 210 person-rem.[72] Holding Union Electric accountable for the 165 person-rem goal for the outage improperly establishes a precedent that would restrict Union Electric's ability to establish challenging management goals for a refueling outage before actual estimates and planning take place.

Whether "job" is defined as a RWP or a WAD, the aggregate of those estimates for Refueling Outage 10 at Callaway was 210 Rem. For determining the significance of an ALARA finding, the SDP relies on determining the total dose and percent increase over estimate of the job associated with the finding. Since, the dose estimates on RWPs are sums of the WADs covered by each RWP, the dose estimate for Refueling Outage 10 would be the same regardless of whether the job is defined at the WAD or RWP level. In September 1999, one month prior to Refueling Outage 10, either of these sums was 210 person-rem.

It is also appropriate, in evaluating Union Electric's ALARA performance during Refueling Outage 10, to adjust the original estimate by 25% for higher dose levels than expected during exposure estimating. Dose rate levels were found during the entry to be 25 to 50% higher than the dose rate levels expected during planning. Using only the lower value of 25% would account for 35 person-rem of the increase over the 210 person-rem budget.[73] The estimate of dose rates during ALARA planning was properly based on past operating experience. Dose rate estimating was not included as an example of poor ALARA practices in the NOV. Therefore, the amount of the exposure increase attributable to higher dose rates should not be considered in determining the significance of the finding.

Emergent work during Refueling Outage 10, such as some RCP seal replacement work, resulted in an increase in scope to the Refueling Outage 10. This added work would account for 13 person-rem of the increase over the 210 person-rem budget.[74] This emergent work should not be considered in determining the significance of the ALARA NOV as it would establish an undesirable precedent that emergent work cannot be added to an outage without deleting other work or risking enforcement action.

After subtracting the exposure attributable to higher dose rates and increased work scope, the remaining excess exposure for Refueling Outage 10 is 47 person-rem (about 25% of the 210 person-rem budget). Excess exposure of about 25% supports Union Electric's analysis that there

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[72] Union Electric presentation at November 9, 2000, regulatory conference, Slide 5.

[73] Union Electric presentation at November 9, 2000, regulatory conference, Slide 5.

[74] Union Electric presentation at November 9, 2000, regulatory conference, Slide 5.

is a strong ALARA program, but there is need for improvement in ALARA execution. As such, assigning the NOV a significance of "no color" rather than three White findings would be appropriate.

C. In the alternative, significance could be based on Union Electric's performance during Refueling Outage 10 by analyzing representative jobs.

In the alternative, even if the NRC decided to evaluate selected jobs from Refueling Outage 10 and use analysis of them as a measure of the significance of the ALARA NOV, the jobs selected were not appropriately analyzed. The NRC assessed the significance of performance on 6 RWPs, which were not causally related to the five examples of ALARA work practices listed in the NOV and which do not reflect that Callaway procedures intend that the WAD be the lowest level of ALARA planning.

The NRC Inspection Manual identifies that exposure returns over 50% of estimate on 5 person-rem jobs is a good screen to focus inspection. The SDP then inappropriately translates these screening criteria into a color significance decision tree that makes the mere existence of such jobs a finding. NRC Inspection Manual chapter 0609-04 defines a finding as any detail noted during an inspection that has been placed in context and initially determined to be of sufficient potential significance to warrant more detailed review using the SDP. In order to be determined of potential significance, a job with over 5 person-rem exposure should have the excess exposure causally related to the violation of ALARA noted during the inspection.

The NRC identified in the NOV five examples of work practices not resulting in ALARA collective doses during Callaway Refueling Outage 10. Any numerical criteria to determine significance should be on jobs associated with those examples. Instead, the NRC significance determination process analyzed four RWPs that involved total exposure over 5 person-rem and 2 RWPs that were over 25 person-rem. As all six RWPs had an increase of over 50% compared to original estimates, the NRC determined the significance as 3 white findings. If the significance had been determined for each RWP based on the amount of the excess exposure attributable to the examples of degraded ALARA work practices noted in the NOV, the conclusion would have been findings with "no color", as follows:

- Planning and conducting maintenance near RCS and steam generator drains earlier in outage than in the past and not filling steam generator secondary sides were examples of degraded ALARA work practices.[75] They were associated with four RWPs. The NRC inspection report notes that this was only a 25% factor on exposure increase.[76]

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75 NRC Notice of Violation EA-00-208, examples a, b, and c.

76 NRC Inspection Report 50-483/00-17 of October 4, 2000, enclosure Section 2OS2.b.

- Conducting insufficient mock-up training was an example of a degraded ALARA work practice.[77] It was associated with three RWPs. The actual fraction of excess exposure attributable to this example is not clear, but it could not have been a 50% factor on exposure.[78]

- Ineffective communications between Union Electric and the primary contractor was an example of a degraded ALARA work practice.[79] It was associated with two RWPs. The actual fraction of excess exposure attributable to this example is not clear, but it could not have been a significant factor on exposure.[80]

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77 NRC Notice of Violation EA-00-208, example d.

78 NRC Inspection Report 50-483/00-17 of October 4, 2000, enclosure Section 2OS2.b, identifies the three RWPs as 99-53321, 99-53323 and 99-53324 where additional mock-up training should have been provided. Only by assuming that all of the growth of manhours for the work could have been avoided by mock-up training would there be a 50% increase in exposure due to insufficient mock-up training. Union Electric, *Refuel 10 ALARA Outage Report (October 2, 1999 to November 5, 1999)* issued June 2000, pages 12 and 13, notes that RWP 99-53321 was for manway cover work which expended 513 manhours compared to 300-350 normally, (about 50%) but also notes that some of this growth was due to response to spreads of contamination. Pages 12 and 14 note that RWP 99-53323 was for eddy current testing and electrosleeving and the manhours for this first large scale application of electrosleeving were difficult to estimate what expected manhour performance should have been. The manhours for eddy current work was about twice previous experience, but this was less than half the work so could not be more than a 50% factor on exposure. Pages 12 and 15 note that RWP 99-53324 was for health physics support and that the manhour growth for this RWP was about 62%, primarily, but not solely due to growth in the steam generator work.

79 NRC Notice of Violation EA-00-208, example e.

80 NRC Inspection Report 50-483/00-17 of October 4, 2000, enclosure Section 2OS2.b, identifies the two RWPs as 99-53324 and 99-53022. Union Electric, *Refuel 10 ALARA Outage Report (October 2, 1999 to November 5, 1999)* issued June 2000, pages 12 and 15 note that RWP 99-53324 was for health physics support, the manhour growth for this RWP was about 62% and, although poor communications were a factor, the increase was primarily due to growth in the steam generator work. Page 16 notes that RWP 99-53022 was for foreign object removal from the steam generator secondary side. The higher exposure was due to higher work scope, as more objects needed to be removed. Although communication systems could be improved, communications during Refueling Outage 10 was better than past experience due to incorporating lessons learned.

- One RWP, reactor coolant pump seal removal and replacement, was not associated with any of the examples in the NOV,[81] but it was still factored in for significance determination.

Thus, an analysis of the exposures relating to the five examples cited in the NOV results in findings with "no color."

D. **Even if use of the SDP for ALARA is considered retroactively applicable to Refueling Outage 10 at Callaway and even if the SDP is applied to jobs without being related either to the scope of the NOV or the issue raised in the NOV, the ALARA SDP metric should be applied to jobs as defined by WADs at Callaway, not to the RWPs.**

The NRC applied the SDP for ALARA at the RWP level as that was thought to be the lowest level where ALARA planning was conducted.[82] Since the NRC NOV was based on examples of allegedly deficient conduct of work, it is an error to determine the significance of the NOV based on the conduct. The controlling factor to define what is a job should be what Union Electric's procedures intend to be the lowest level. The procedures, which were not criticized in the NRC inspection report, intend the lowest level of ALARA planning to be the WAD.

Under this analysis, Union Electric considers that the proper result should be one finding of "no color." In evaluating this contention, the NRC should consider the following ways in which the ALARA SDP overstates the significance of the finding:

- High three-year rolling average is only indicative of possible occupational exposure control issues and not conclusive proof. As discussed above, the reasons that Callaway's three-year rolling exposure average exceeds other PWR's is due in large part to plant design issues and the maintenance strategies selected. NRC IM Chapter 0609, attachment 2 asks the question whether the three-year rolling average collective dose is above average. If the answer was tempered with judgment, the conclusion should be, "yes, but not above 'average' for a plant with axial offset anomaly and old Westinghouse

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81 NRC Inspection Report 50-483/00-17 of October 4, 2000, enclosure Section 2OS2.b, identifies that due to an emergent change to the scope, all reactor coolant pump seals could not be worked with the steam generator's secondary sides full as originally planned. Workers moved tooling between pumps multiple times as other work allowed seal work to proceed. Neither of these points is identified as poor ALARA work practices in the NRC Notice of Violation. Arguably, the Inspection Report is not listing deficiencies, but only documenting the complexity associated with accomplishing multiple jobs to ensure the integrity of the Reactor Coolant Boundary.

82 NRC letter to Union Electric of January 9, 2001.

Inconel 600 tubes in its steam generators". The result under the attachment, with no other considerations, would then be that the finding has "no color".

- Callaway has a strong ALARA program, to identify degraded performance and establish corrective actions for exposure control, as shown by strong upchecks from peer reviews and past NRC inspections. If the SDP considered the mitigating value of licensee corrective actions, like the enforcement policy itself does,[83] then the improvement actions initiated by Union Electric should reduce the significance associated with the NOV.

- Previously, the only consequence attached to over aggressively low dose estimates was economic, as it would lead the licensee to conduct additional investigations for poor ALARA work practices. It overstates the significance determination of the NOV to not take into account that Union Electric, during Refueling Outage 10, made a conscious decision to spend additional effort on investigating the causes of dose overruns and not on correcting the unrealistic dose estimates.

If "job" is defined at the WAD (not RWP) level at Callaway, then there are five jobs that exceeded 5 person-rem, but none that exceeded 25 person-rem.[84] As discussed in the R. D. Affolter letter dated November 16, 2000, the significance attributable to having five 5 person-rem jobs exceed their dose estimates by over 50% is one White finding. However, after careful reflection on the implementation of the ALARA SDP, Union Electric now believes that a "no color" finding is appropriate. The different ways of looking at the same fact pattern and of concluding different enforcement significance highlight the problems with the SDP metric as adopted.

III. THE USE OF THE SDP FOR ALARA SHOULD BE SUSPENDED UNTIL IT CAN BE REVISED TO BE CONSISTENT WITH GOALS OF THE REVISED ROP AND WITH GOALS OF ALARA

SECY-00-0049 noted that the feed-back from the pilot programs to implement the revised ROP indicates that further experience is needed with the revised process. It also noted that the fire protection SDP needed to be revised to reduce its complexity and improve its usability prior to initial implementation. Consistent with the action taken for the fire protection SDP, the NRC should suspend use of the ALARA SDP until improved determination methodology for ALARA findings can be developed. The current ALARA SDP produces significance determinations that overstate the significance of the findings and do not account for appropriate consideration of licensee identification and mitigation actions.

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83 *See*, SECY-00-0061, page 2, which discusses mitigation discretion under the Enforcement Policy and notes that the Enforcement Policy has been modified so that mitigation discretion for some circumstances does not normally apply to violations associated with issues evaluated by the SDP.

84 Letter dated November 16, 2000, from R.D. Affolter, Vice President, Nuclear, Union Electric Company, to the U.S. Nuclear Regulatory Commission (ULNRC-4343).

ALARA has operated successfully for 30 years. The ALARA SDP now creates new criteria, which can lead to White findings at almost every ALARA inspection. This result will not lead to increased focus on significant safety issues as intended by the new ROP, but rather, the focus will be ALARA findings. Since there is no screening of significance in the ALARA SDP for whether there is a risk of exceeding individual exposure limits, the revised ROP may result in focus on ALARA findings of no health or safety significance. Collective exposures criteria of 5 or 25 Rem per job and three-year rolling averages for PWR and BWR are acceptable screens for the NRC inspection procedures to focus reviews but should not be used to define the safety significance of ALARA findings. Also, the ALARA SDP does not provide mitigation of significance for licensee self-determination or prompt and effective corrective actions. Since the collective exposure results are only one potential indication of how the licensee's ALARA process is working, effective action to prevent recurrence should be a key part of determining the significance of an ALARA issue.

A fatal flaw in the ALARA SDP is that it assumes a healthy ALARA program. If dose estimates are invalid for a job, the SDP provides no correct way to assess significance of findings. Significance based on planning performance, which although key, is only one of numerous aspects of ALARA program inspection procedure. By overstating the importance of one element of the ALARA program, the SDP threatens to force licensee's ALARA programs out of balance.

The use of the SDP for ALARA should be suspended as it is fatally flawed. Union Electric understands that the SDP for ALARA resulted from a well-intentioned attempt to establish a metric for inspection of the ALARA programs at nuclear plants. The new Regulatory Oversight Process has made many needed improvements to the inspection and enforcement at nuclear plants and Union Electric strongly supports the effort. Union Electric's experience with the SDP for ALARA, however, demonstrates that it is inconsistent with the risk-informed basis of the Regulatory Oversight Process and counter-productive to the intent of ALARA. Consistent with suspending the use of the SDP for ALARA until lessons learned are incorporated, the significance of the ALARA NOV issued to Callaway should be assessed as "no color."

Issue Date: 4/24/00

NRC INSPECTION MANUAL IIPB

Manual Chapter 0305

OPERATING REACTOR ASSESSMENT PROGRAM

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0305-01 PURPOSE

The Revised Reactor Oversight Process is the result of an effort by the NRC to improve the NRC's inspection, assessment, and enforcement programs. The result is a regulatory framework (exhibit 1) that is more objective, understandable, and predictable and focuses agency resources on areas that have the greatest impact on safe plant operation. The Operating Reactor Assessment Program evaluates the overall safety performance of operating commercial nuclear reactors and communicates those results to licensee management, members of the public, and other government agencies.

The assessment program (exhibit 2) collects information from the inspection program and performance indicators in order to enable the agency to arrive at objective conclusions about the licensee's safety performance. Based on this assessment information, the process determines the appropriate level of agency response including supplemental inspection, demands for information, confirmation of specific corrective actions, or orders, up to and including a plant shutdown. The assessment information and agency response are then communicated to the public. Follow-up agency actions, as applicable, are conducted to ensure that the corrective actions designed to address performance weaknesses were effective.

0305-02 OBJECTIVES

02.01 To collect information from inspection findings and performance indicators.

02.02 To arrive at an objective assessment of licensee safety performance using performance indicators and inspection findings.

02.03 To assist NRC management in making timely and predictable decisions regarding appropriate agency actions used to oversee, inspect, and assess licensee performance.

02.04 To provide a method for informing the public and soliciting stakeholder feedback on the NRC's assessment of licensee performance.

02.05 To provide a process to follow up on areas of concern.

0305-03 APPLICABILITY

This manual chapter applies to all operating commercial nuclear reactors except those sites that are under IMC 0350, "Staff Guidelines For Assessment and Review of Plants That Are Not Under The Routine Reactor Oversight Process". The contents of this manual chapter do not restrict the NRC from taking any necessary actions to fulfill its responsibilities under the Atomic Energy Act of 1954 (as amended).

0305-04 DEFINITIONS

04.01 Significance Determination Process (SDP). A risk characterization process that is applied to inspection findings such that the overall licensee performance assessment process can compare and evaluate the findings on a significance scale similar to the performance indicators.

04.02 Degraded Cornerstone. A cornerstone that has two or more white inputs or one yellow input.

04.03 Repetitive Degraded Cornerstone. A cornerstone that is degraded (2 white inputs or 1 yellow input) for five or more consecutive quarters.

04.04 Multiple Degraded Cornerstones. Two or more cornerstones are degraded in any one quarter.

04.05 Inspection Finding. As used in IMC 0610* "Inspection Reports", an observation that has been placed in context. Findings are assigned a color based on their risk significance as an outcome of the significance determination process. Listed below are the colors associated the risk significance of these findings:

Green Findings - Issues that, while not desirable, represent very low safety significance.

White Findings - Issues with low to moderate safety significance.

Yellow Findings - Issues with substantial safety significance which would require the NRC to take additional actions.

Red Findings - Issues with high safety significance and an unacceptable loss of safety margin which would result in the NRC taking significant actions that could include ordering the plant to be shutdown.

04.06 Assessment Period. A rolling 12 month period that contains 4 quarters of performance indicators and inspection findings.

Note: An inspection finding is normally carried forward in the assessment process for a total of four calender quarters. However, the inspection finding will not be removed from consideration of future agency actions (per the Action Matrix) until the identified weaknesses in the root cause evaluation have been corrected.

04.07 Annual Assessment Cycle. The 12 month assessment period, April 1 through March 31, that culminates in a Commission briefing.

04.08 Assessment Inputs. As used in this manual chapter, assessment inputs are the combination of performance indicators and inspection findings for a particular plant that are combined in the assessment process in order to determine appropriate agency actions.

04.09 MC 0350 Process. As used in this manual chapter, an oversight process that oversees licensee performance, inspections, and restart efforts for plants with significant performance problems.

04.10 Safety-Conscious Work Environment. An environment in which employees feel free to raise safety concerns, both to their management and to the NRC, without fear of retaliation.

0305-05 RESPONSIBILITIES AND AUTHORITIES

05.01 Executive Director for Operations (EDO)

a. Oversees the activities described in this manual chapter.

b. Approves deviations from the Multiple/Repetitive Degraded Cornerstone column of the Action Matrix.

05.02 Director, Office of Nuclear Reactor Regulation (NRR)

a. Implements the requirements of this manual chapter within NRR.

b. Develops assessment program policies and procedures.

c. Ensures uniform program implementation and effectiveness.

d. Concurs on all agency actions that deviate from the Regulatory Response and Degraded Cornerstone columns of the Action Matrix as described in section 06.01.e of this manual chapter.

05.03 Regional Administrators

a. Implements the requirements of this manual chapter within their respective regions.

b. Develops and issues Annual Assessment Letters to each licensee, which contain a concise assessment of licensee performance using information captured by performance indicators and NRC inspection findings.

c. Directs allocation of inspection resources within the regional office based on the Action Matrix.

d. Establishes a schedule and determines a suitable location for the annual public meeting with each licensee to ensure a mutual understanding of the issues discussed in the Annual Assessment Letter.

e. Suspends the end-of-year performance review for those plants that have been transferred to the Inspection Manual Chapter 0350 process.

f. Approves agency actions that deviate from the Regulatory Response and Degraded Cornerstone columns of the Action Matrix as described in section 06.01.e of this manual chapter.

g. Recommends deviations from the Multiple/Repetitive Degraded Cornerstone column of the Action Matrix.

05.04 Chief, Inspection Program Branch

a. Develops program guidance.

b. Collects feedback from the regional offices and assesses execution of the Operating Reactor Assessment Program to ensure consistent application.

c. Recommends and implements improvements to the Operating Reactor Assessment Program.

0305-06 BASIC REQUIREMENTS

06.01 Assessment Process

Licensee performance is reviewed over a 12-month period through the reactor assessment process (exhibits 3 and 4). The assessment process consists of a series of reviews which are described below.

Each regional office will conduct an ongoing review of the performance of their assigned plants. Inspections are conducted on a continuous basis in accordance with IMC 2515 and performance indicators are reported quarterly by the licensee. Assessment activities occur at quarterly intervals. Resident inspectors and branch chiefs shall maintain a continuous awareness of plant performance. If an inspection finding is identified during the quarter that is risk significant (i.e. greater than green) the regional office may address this issue without waiting until the end of the quarter, if appropriate. With respect to performance indicators, there is no intention that performance indicators be monitored on a real time basis. However, the regional office may take the appropriate action if the licensee contacts the regional office regarding a performance indicator that will definitively cross the green/white threshold at the end of the quarter. Additionally, the agency will not wait until the annual Agency Action Review meeting to address plants with significant performance problems. Plants with significant performance problems are those plants that are in the Multiple/Repetitive Degraded Cornerstone column or the Unacceptable Performance column of the Action Matrix.

The inspectors will normally use the SDP to evaluate inspection findings. However, the NRC enforcement policy also describes violations which the SDP process can not evaluate for risk significance (i.e., violations that involve actual safety significance, impede the regulatory process, or involve willfulness). This aspect of the enforcement policy shall be followed for violations outside of the SDP process. Regional management should notify the licensee in writing if additional inspection activities are scheduled to occur within the current quarter via an Assessment Follow-Up Letter (exhibit 7).

a. Quarterly Review. The quarterly review utilizes PI data submitted by licensees and inspection findings compiled over the previous twelve months (which includes three new months of assessment inputs). This review will be conducted after the conclusion of each quarter during the annual assessment cycle. The regional office will review these results to determine appropriate agency actions per the Action Matrix. The most recent performance indicators and inspection findings shall be considered in determining agency action. This may include previous inspection findings as these findings are normally carried forward in the assessment process for four consecutive quarters.

The responsible DRP Branch Chief will review the most recently submitted PIs (which should be submitted 21 days after the end of the quarter) and the inspection findings contained in the plant issue matrix (PIM) to identify any changes in performance trends. The review should be completed within five weeks of the end of the quarter. The BC shall utilize the Action Matrix to identify the potential scope of NRC actions not already embedded in the existing inspection plan. The regional office will notify the licensee via an Assessment Follow-Up Letter when assessment input thresholds are crossed. The Assessment Follow-Up Letter should be issued within two weeks of completing the quarterly review, if applicable. The regional office should still perform the supplemental inspection procedure even if a performance indicator re-enters the green band.

Additionally, for plants whose performance is in the Multiple/Degraded Cornerstone column of the Action Matrix consideration shall be given at each quarterly review for engaging senior licensee and agency management in discussions associated with 1) transferring the plant to the IMC 0350 process and 2) declaring licensee performance to be unacceptable in accordance with the guidance contained within this manual chapter.

Note: If the agency determines that a licensee's performance is unacceptable then a shutdown order will be issued.

b. Mid-Cycle Review. The mid-cycle review utilizes the most recent performance indicators and inspection findings compiled over the previous twelve months. This review incorporates activities from the quarterly review after the conclusion of the second quarter of the annual assessment cycle. The output of this review is a Mid-Cycle Letter (exhibit 8) instead of an Assessment Follow-Up Letter. Additional activities include planning inspection activities for the next twelve months as well as discussing any insights into potential cross-cutting issues (problem identification and resolution, human performance, and safety-conscious work environment).

c. A Mid-Cycle Review Meeting. Will be chaired by a Division of Reactor Projects (DRP) or Division of Reactor Safety (DRS) Division Director (DD). The DRP Branch Chiefs responsible for their plants should take the lead in presenting the overall results of the review to the Division Director. The DRS Branch Chiefs shall coordinate with the appropriate DRP Branch Chiefs to provide adequate support for the presentation and the development of the inspection plan. Other participants shall include applicable regional and resident inspectors, a Senior Reactor Analyst, a representative from the Inspection Program Branch (IIPB), the regional Allegations Coordinator or the Agency Allegations Advisor, and any other additional resources deemed necessary by the regional offices. The Action Matrix will be used to determine the scope of agency actions in response to the assessment inputs. The Mid-Cycle Review will be completed within six weeks of the end of the second quarter of the end of the annual assessment cycle.

The outputs of the mid-cycle review is a Mid-Cycle Letter (exhibit 8) and shall be issued within three weeks of the completion of the mid-cycle review. This letter shall contain:

1. A summary of performance indicators and inspection findings that were outside of the licensee response band (including any associated cross-cutting issues) for the most recent quarter as well as discussion of previous action taken by the licensee and the agency. Performance issues from previous quarters may be discussed if:
   (a) The agency's response to an issue had not been adequately captured in previous correspondence to the licensee.
   (b) These issues, when combined with assessment inputs from the most recent quarter, result in increased regulatory action per the Action Matrix that would not be apparent from reviewing only the most recent quarter's results.
2. A qualitative discussion of distinct adverse trends as indicated by substantial cross-cutting issues that have not resulted in performance indicators or inspection findings outside of the licensee response band. Safety-conscious work environment issues shall only be discussed if the agency has previously engaged the licensee via a meeting or correspondence regarding a potential or actual "chilled work environment".
3. A statement of any actions, beyond the baseline inspection program, to be taken by the agency as well as any actions previously taken by the licensee.
4. An inspection plan for the next twelve months that will be updated (as necessary) at the End-of-Cycle Review meeting.

d. End-of-Cycle Review. The End-of Cycle Review is a comprehensive assessment of licensee performance using the most recent performance indicators and inspection findings from the previous 12 months. This review incorporates activities from the quarterly review after the conclusion of the annual assessment cycle. The output of this review is an Annual Assessment Letter (exhibits 9,10,11, and 12) instead of an Assessment Follow-Up Letter. Additional activities include planning inspection activities for the next twelve months, discussing any insights into cross-cutting issues (problem identification and resolution, human performance, and safety-conscious work environment), and providing an input into the Agency Action Review Meeting.

The End-of-Cycle Review Meeting will be chaired by the Regional Administrator or his/her designee. The DRP and DRS Division Directors (or designees) will present the results of the annual review. The Director of NRR (or another member of the Executive Team) should attend the meeting to provide the program office's perspective. Other participants should include DRP and DRS Branch Chiefs, Senior Reactor Analysts (SRAs), a representative from the Inspection Program Branch (IIPB), the regional Allegations Coordinator or the Agency Allegations Advisor, and senior representatives from the Division of Licensing Project Management, Office of Investigations, Office of Enforcement, and Office of Research. The End-Of-Cycle meeting should be held within six weeks of the end of the assessment cycle. The Action Matrix will be used to determine the scope of agency actions in response to assessment inputs.

The output of the End-of-Cycle Review is the Annual Assessment Letter (exhibits 9,10,11,and 12). This letter will be issued within one week after the Agency Action Review meeting and shall contain the following:

1. A statement regarding overall plant performance based on the most recent performance indicators and the previous 12 months of inspection findings.
2. A summary of any PIs or inspection findings that are currently outside of the licensee response band including a discussion of followup action taken by the licensee and the agency.
3. A brief summary of licensee performance that had been outside of the licensee response band for the first three quarters of the assessment cycle.
4. A qualitative discussion of adverse trends as indicated by substantial cross-cutting issues that have not resulted in performance indicators or inspection findings outside of the licensee response band. Safety-conscious work environment issues shall be discussed only if the agency has previously engaged the licensee via a meeting or correspondence regarding a potential or actual "chilled work environment".
5. A statement of any actions, beyond the baseline inspection program, to be taken by the agency as well as any actions previously taken by the licensee.

e. Agency Action Review. An Agency Action Review Meeting is conducted approximately two weeks after the End-of-Cycle Review by senior NRC managers and is chaired by the Executive Director for Operations (EDO) or designee. This review uses data compiled during the End-of-Cycle review and involves a collegial review by senior NRC managers and staff of the appropriateness of agency actions for plants with significant performance issues, overall industry performance, and the results of the oversight process self-assessment. Plants with significant performance weaknesses are those plants that are in the Multiple/Repetitive Degraded Cornerstone or Unacceptable performance column of the Action Matrix.

The Regional Administrators and the Director of NRR will brief the participants on overall industry performance, oversight process self-assessment results, and any plants with significant performance weaknesses as determined by the Action Matrix. The Agency Allegations Advisor, senior representative(s) from the Office of Nuclear Material Safety and Safeguards (NMSS), Office of Investigations, Office of Enforcement, Office of Research, Office of Public Affairs, Office of General Counsel, Office of the Chief Financial Officer, and Office of the Chief Information Officer will attend the meeting. All of the Annual Assessment Letters (exhibits 9,10,11, and 12) shall be sent to the licensee no later than one week after completing the Agency Action Review meeting to ensure that the annual assessment letters are publicly available prior to the Commission meeting.

f. Commission Meeting. Annually the EDO will brief the Commission to convey the results of the Agency Action Review Meeting to the Commission. The Commission should be briefed within eleven weeks of the end of the assessment cycle.

g. Action Matrix. The Action Matrix (exhibit 5) was developed with the philosophy that, within a certain level of safety performance (i.e., the licensee response band), the licensee should be allowed to address their performance issues. Agency action beyond the baseline inspection programs should occur only if assessment input thresholds are exceeded. The Action Matrix identifies the range of NRC and licensee actions and the appropriate level of communication for varying levels of licensee performance. The Action Matrix describes a graded approach in addressing performance issues. A few terms are used throughout the discussion of the Action Matrix. These are:

- ***Regulatory Performance Meetings.*** Regulatory performance meetings are held between licensees and the agency to discuss risk significant performance issues (i.e., outside of the licensee response band) that resulted in licensee performance outside of the licensee response band. Each risk significant assessment input shall be discussed in one of the forums listed below in order to arrive at a shared understanding of the performance issues, underlying causes, and planned licensee actions. These meetings may take place at a regulatory conference, periodic inspection exit meetings between the agency and the licensee, or public meetings. This meeting should be documented in an inspection report, a public meeting summary, or conference call minutes.
- *Licensee Action.* Anticipated actions by the licensee in response to the performance described in the appropriate column of the Action Matrix. If these actions are not being taken by the licensee then the agency may expand the scope of the applicable supplemental inspection to appropriately address the area(s) of concern. This would not be considered a deviation from the Action Matrix in accordance with section 06.01.e of this manual chapter.
- ***NRC inspection.*** The range of NRC inspection activities in response to the performance described in the appropriate column of the Action Matrix.
- ***Regulatory actions.*** Range of actions to be taken by the agency to in response to the performance described in the appropriate column of the Action Matrix.

Below is a discussion of the components of the Action Matrix. Refer to exhibit 5 for a depiction of the Action Matrix.

1. **Response**

The Action Matrix lists expected NRC and licensee actions based on the inputs to the assessment process. Actions are graded such that the agency becomes more engaged as licensee performance declines. Listed below are the range of expected NRC and licensee actions for each column of the Action Matrix:

- **Licensee Response Column** - All assessment inputs are green. The licensee will receive only the baseline inspection program and identified deficiencies will be placed into the licensee's corrective action program.
- **Regulatory Response Column** - Assessment inputs result in one or two white inputs in different cornerstones. The licensee is expected to place the identified deficiencies in its corrective action program and perform an evaluation of the root and contributing causes. The licensee's evaluation will be reviewed during inspection procedure 95001 *Supplemental Inspection for One or Two White Inputs in a Strategic Performance Area.* Following completion of the inspection, the Branch Chief or Division Director should discuss the performance deficiencies and the licensee's proposed corrective actions with the licensee. The regulatory performance meeting will normally occur at an inspection exit meeting or a conference call between the licensee and the appropriate Branch Chief (or Division Director).

- **Degraded Cornerstone Column** - Assessment inputs result in a degraded cornerstone or 3 white inputs to any Strategic Performance Area. The licensee is expected to place the identified deficiencies in its corrective action program and perform an evaluation of the root and contributing causes for both the individual and the collective issues. The licensee's evaluation will be reviewed during inspection procedure 95002 *Supplemental Inspection For One Degraded Cornerstone Or Any Three White Inputs in a Strategic Performance Area.* Also, an independent assessment of the extent of condition will be performed by the region using appropriate inspection procedures chosen from the tables contained in Appendix B to Inspection Manual Chapter 2515. Following completion of the inspection, the Division Director or Regional Administrator should discuss the performance deficiencies and the licensee's proposed corrective actions with the licensee. The regulatory performance meeting will normally consist of a public meeting between the licensee and the appropriate Division Director (or Regional Administrator).
- **Multiple/Repetitive Degraded Cornerstone Column** -Assessment inputs result in a repetitive degraded cornerstone, multiple degraded cornerstones, multiple yellow inputs or a red input. The licensee is expected to place the identified deficiencies in its corrective action program and perform an evaluation of the root and contributing causes for both the individual and the collective issues. This evaluation may consist of a third party assessment. Inspection procedure 95003 *Supplemental Inspection for Repetitive Degraded Cornerstones. Multiple degraded Cornerstones, Multiple Yellow Inputs, or One Red Input* will be performed to determine the breadth and depth of the performance deficiencies. Following the completion of the inspection, the EDO or his designee, in conjunction with the Regional Administrator and the Director of NRR, will decide whether additional agency actions are warranted. These actions could include additional demands for information, confirmation of specific corrective actions, or orders, up to and including a plant shutdown. The regulatory performance meeting will normally consist of a public meeting between the licensee and the Regional Administrator (or Executive Director of Operations).
- Unacceptable Performance Column - Licensee performance is unacceptable and continued plant operation is not permitted within this column. In general, it is expected that entry into the multiple/repetitive degraded cornerstone column of the Action Matrix will precede plant consideration in the Unacceptable Performance Column. The Commission will meet with senior licensee management in a regulatory performance meeting to discuss the licensee's degraded performance and the corrective actions which will need to be taken before operation of the facility can be resumed. The NRC oversight of plant performance will also be placed under the guidance of IMC 0350. Unacceptable performance represents situations in which the NRC lacks reasonable assurance that the licensee can or will conduct its activities without undue risk to public health and safety. Examples of unacceptable performance may include:

- Multiple significant violations of the facility's license, technical specifications, regulations, or orders.
- Loss of confidence in the licensee's ability to maintain and operate the facility in accordance with the design basis (e.g., multiple safety significant examples where the facility was determined to be outside of its design basis, either due to inappropriate modifications, the unavailability of design basis information, inadequate configuration management, or the demonstrated lack of an effective problem identification and resolution program).

- A pattern of failure of licensee management controls to effectively address previous significant concerns to prevent their recurrence.

> Note: If the agency determines that a licensee's performance is unacceptable then a shutdown order will be issued.

2. **Communication**
Communication between the licensee and the NRC is based on a graded approach. For declining licensee performance, higher levels of agency management will review and sign the assessment letters and conduct the annual public meeting.
3. **Supplemental inspection for a single white issue**
The regional office may elect not to conduct a supplemental inspection for a white finding that was identified as part of a licensee self-assessment activity. In deciding whether to exercise this option, the region should consider the results of past reviews of the licensee's problem identification and resolution program, specifically with regard to the effectiveness of previously performed root cause analyses. The DRP or DRS Division Director will authorize this option with the concurrence of the Inspection Program Branch Chief and should document the basis for the decision not to perform the supplemental inspection in an assessment follow-up letter to the licensee. This is not considered a deviation from the Action Matrix in accordance with section 06.01.e of this manual chapter.

   The purpose of this option is to provide an incentive for a licensees to aggressively pursue the identification and resolution of their own issues.
4. **"Double-Counting"of performance indicators and inspection findings**Some singular events may result in a simultaneous tripping of a performance indicator and an inspection finding. This would appear to result in two assessment inputs combining to cause increased regulatory action per the Action Matrix. For example, two white assessment inputs in the mitigating systems cornerstone would result in increased regulatory action per the degraded cornerstone column of the Action Matrix.

   Singular events should not be "double-counted" in the assessment program. However, the most conservative color from the performance indicator and the inspection finding (i.e. yellow vs. white) shall be used to determine the appropriate agency action according to the Action Matrix. This is not considered a deviation from the Action Matrix as defined in section 06.01.e of this manual chapter.
5. **Timeframe for "counting" inspection findings in the assessment program**The date used for consideration in the assessment program is the date of occurrence for events or the date of the end of the pertinent inspection period for inspection findings. After final determination of the significance of an inspection finding the regional office shall refer back to the appropriate date discussed above to determine if any additional action would have been taken had the significance of the inspection finding been known at that time.

   For example, the performance indicator for Unplanned Scrams was white (low to moderate risk significance) for the second quarter of the assessment cycle. Additionally, there was an inspection finding from the second quarter of the assessment cycle whose final risk significance was determined to be white (low to moderate risk significance) in the third quarter of the assessment cycle. In this case, the appropriate action would be to perform supplemental inspection procedure 950002 vice 95001 which would be documented in the Assessment Follow-Up Letter.

h. <u>Deviations from the Action Matrix</u>. There may be rare instances in which the actions dictated by the Action Matrix may not be appropriate. In these instances, the agency may deviate from the Action Matrix (which is described in section 06.01.d of this manual chapter) to either increase or decrease agency action. A deviation is defined as any actions taken that are inconsistent with the range of actions discussed in section 06.01.d of this manual chapter. A deviation from the Action Matrix requires the appropriate level of senior agency management approval and concurrence. The agency manager responsible for approval of the assessment letter one column to the right of where the licensee's performance is relative to the Action Matrix shall authorize the deviation. For example, if the agency will deviate from the Regulatory Response column of the Action Matrix, the appropriate approval level would be the Regional Administrator with the concurrence of the Director of NRR. Deviations from the Action Matrix shall be documented in the appropriate letter to the licensee (i.e. assessment follow-up letter, mid-cycle or annual assessment letter). The Executive Director for Operations shall authorize proposed deviations from the Multiple/Repetitive Degraded Cornerstone column of the Action Matrix.

Any deviations from the Action Matrix shall be documented in an annual report to the Commission.

i. Relationship with the IMC 0350 Process and Unacceptable Performance. The normal criteria for considering a plant for the IMC 0350 process is 1) plant performance is in the Multiple/Repetitive Degraded Cornerstone column or the Unacceptable Performance column of the Action Matrix, (2) the plant is shutdown (whether voluntary or via an agency order to shutdown), and (3) an agency management decision is made to place the plant in the IMC 0350 process as discussed in IMC 0350. At this point, periodic assessment (quarterly, mid-cycle, and end-of-cycle) of licensee performance is no longer under the auspices of this manual chapter but is now under the IMC 0350 process. This process is more completely described in IMC 0350.

The normal criteria for declaring licensee performance to be unacceptable is 1) plant performance is in the Multiple/ Repetitive Degraded cornerstone column of the Action Matrix **and** 2) the criteria for the Unacceptable Performance column of the Action Matrix as described in section 06.01.d of this manual chapter.

The following are examples of the appropriate level of regulatory engagement between the agency and licensees once a plant has entered the Multiple/Repetitive Degraded Cornerstone column of the Action Matrix:

1. Plant A continues to operate and regulatory engagement is dictated by the Multiple/Repetitive Degraded Cornerstone column of the Action Matrix. The agency performs supplemental inspection procedure 95003 (if not already performed) and the plant remains under the level of oversight dictated by this manual chapter and is not transferred to the IMC 0350 process.
2. Plant B performs a voluntary shutdown to address performance issues. The agency performs supplemental inspection procedure 95003 (if not already performed) and issues a confirmatory action letter to document licensee commitments to the agency. The plant remains under the level of oversight dictated by this manual chapter and is not transferred to IMC 0350 process.
3. Plant C performs a voluntary shutdown to address performance issues. The entry conditions for IMC 0350 have been met and agency management determines that this process should be implemented using the criteria in IMC 0350. At this point, periodic assessment of licensee performance is no longer dictated by this manual chapter and is transferred to the IMC 0350 process. Plant performance is not determined to be unacceptable.

4. Plant D voluntarily shuts down to address performance issues. The agency determines that one of the criteria in paragraph 6.B.1 for unacceptable performance is met. The plant is considered to be in the Unacceptable Performance column of the Action Matrix and a shutdown order is issued by the agency. The plant is transferred to the IMC 0350 process.
5. Plant E is issued an order by the agency to shutdown. The licensee's performance is declared to be unacceptable and the plant will be transferred to IMC 0350.

j. Annual Meeting with Licensee

1. **Scheduling**

   A public meeting with the licensee will be scheduled within 16 weeks of the end of the assessment period to discuss the results of the NRC's annual assessment of the licensee's performance. The 16 week requirement may occasionally be exceeded to accommodate the licensee's schedule. The meeting will be conducted onsite or in the vicinity of the site so that it will be accessible to members of the public. NRC management, as specified in the Action Matrix, will conduct the public meeting.

2. **Meeting Preparation**

   The region shall notify those on distribution for the annual assessment letters of the meeting with the licensee. The region shall notify the media and State and local government officials of the issuance of the annual assessment letter and of the meeting with the licensee. Adequate notification of the meeting will be accomplished by distribution within at least 10 working days to the Public Document Room of the letter scheduling the meeting with the licensee.

3. **Conduct of Licensee Meeting**

   The annual public meeting is intended to provide a forum for a candid discussion of issues related to the licensee's performance. NRC management, as specified in the Action Matrix, will discuss the agency's evaluation of licensee performance as documented in the annual assessment letter. The licensee should be given the opportunity to respond at the meeting to any information contained in the Annual Assessment Letter.

   The annual meeting will be a public meeting. The meeting must be closed for such portions which may involve matters that should not be publicly disclosed under Section 2.790 of Title 10 of the Code of Federal Regulations (10 CFR 2.790). Members of the public, the press, and government officials from other agencies should be treated as observers during the conduct of the meeting. Attendees should be given the opportunity to ask questions of the NRC representatives at the conclusion of the meeting.

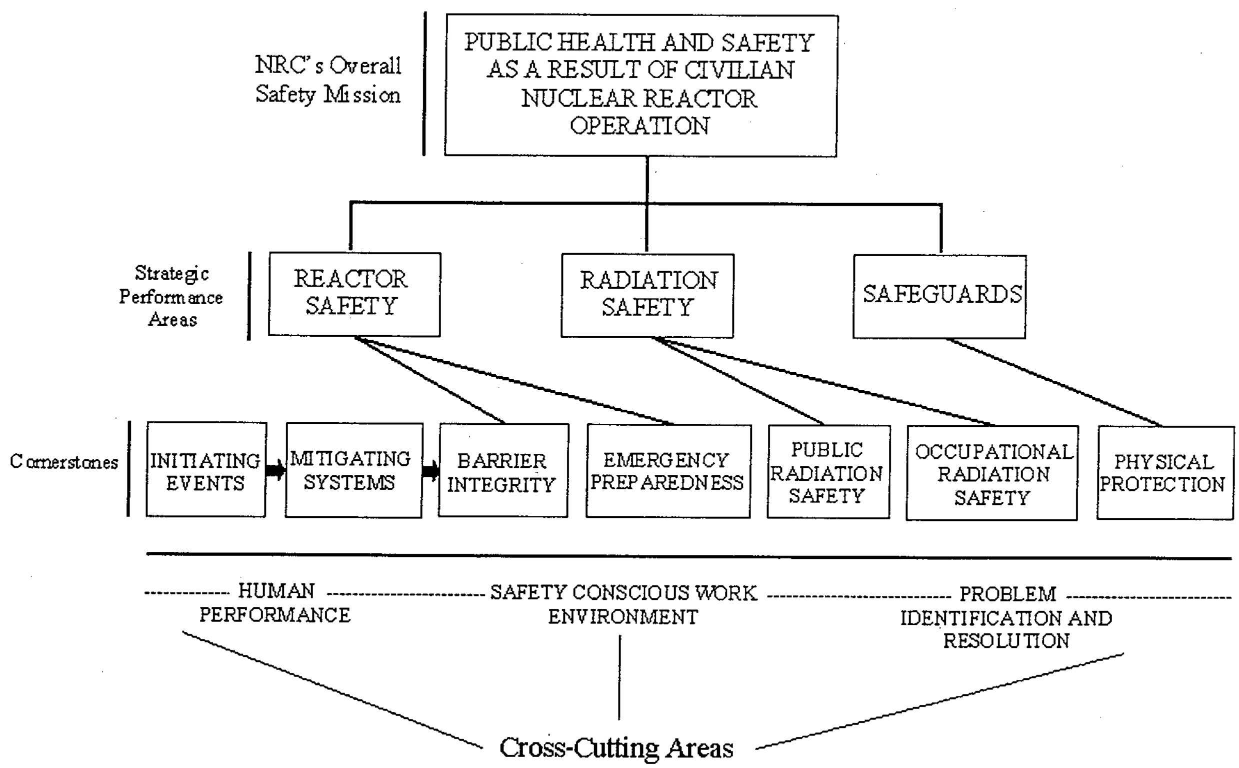
END

Exhibits:

1. Regulatory Framework
2. Reactor Oversight Process
3. Process Activities
4. Schedule of events during annual assessment cycle
5. Action Matrix
6. Plant X 4Q/1999 Performance Summary
7. Sample Assessment Follow-up Letter
8. Sample Mid-Cycle Letter
9. Sample Annual Assessment Letter For Plants in the Licensee Response Column

10. Sample Annual Assessment Letter for Plants in the Regulatory Response Column
11. Sample Annual Assessment Letter for Plants in the Degraded Cornerstone Column
12. Sample Annual Assessment Letter for Plants in the Multiple/Repetitive Degraded Cornerstone Column

Exhibit 1: REGULATORY FRAMEWORK



OCRV
AUDIT OBSERVA

Audit No. ____________

Log No. ____________

Name ____________ Orga

Requirement Reference ____________

Question/Concern: ____________

Response: ____________

Cleared for Submittal to Affected Organization

Exhibit 3 - Process Activities

Level of Review	Frequency/Timing	Participants (* indicates chairperson)	Desired Outcome	Communication
Continuous	Continuous	SRI*, RI, regional inspectors, analysts	Performance awareness	None required, notify licensee by an Assessment Follow-Up letter only if thresholds crossed
Quarterly	Once per quarter/ Five weeks after end of quarter	DRP: BC*, PE, SRI, RI	Input/verify PI/PIM data, detect early trends	Update data set, notify licensee by an Assessment Follow-Up letter only if thresholds crossed
Mid-Cycle	At mid-cycle/ Six weeks after end of second quarter	Divisions of Reactor Safety (DRS) or DRP DD*, DRP and DRS BCs	Detect trends, plan inspection	Mid-cycle letter with an 12-month inspection plan
End-of-Cycle	At end-of-cycle/ Six weeks after end of assessment cycle	DRS or DRP DD, RAs*, NRR representative, BCs, principal inspectors. SRAs	Assessment of plant performance, oversight and coordination of regional actions	Annual Assessment Letter with an 12-month inspection plan
Agency Action Review	Annually/ Two weeks after end-of-cycle review	EDO*,DIR NRR, RAs, DRS/DRP DDs, IIPB, OE, OI, other HQ offices as appropriate	Oversight and coordination of agency-level actions	Commission briefing, followed by public meetings with individual licensees to discuss assessment results

Exhibit 4 - Schedule of events during the annual assessment cycle

Event	Date	Note
Beginning of full implementation of Revised Reactor Oversight Process	04/02/00	N/A
1) End of first quarter of assessment cycle 2) End of inspection cycle	06/30/00	N/A
First quarter PI data available internally	07/21/00	3 weeks after end of first quarter
First Quarter review completed	08/04/00*	5 weeks after end of first quarter
1) End of second quarter of assessment cycle 2) End of inspection cycle	09/30/00	N/A
Second quarter PI data available internally	10/21/00	3 weeks after end of second quarter
Mid-cycle review completed	11/11/00*	6 weeks after end of second quarter
Mid-Cycle letters sent to licensees	12/02/00*	3 weeks after completion of mid-cycle review
1) End of third quarter of assessment cycle 2) End of inspection cycle	12/31/00	N/A
Third quarter PI data available internally	01/21/01	3 weeks after end of third quarter
Third Quarter review completed	02/04/01*	5 weeks after end of third quarter
1) End of fourth quarter of assessment cycle 2) End of inspection cycle	03/31/01	N/A
Fourth quarter PI data available internally	04/21/01	3 weeks after end of fourth quarter
End-of-Cycle review completed	05/12/01*	6 weeks after end of fourth quarter
Agency Action Review meeting completed	05/26/01*	2 weeks after completion of end-of-cycle review
Annual assessment letters sent out to licensees	06/02/01*	1 week after completion of Agency Action Review meeting
Commission meeting completed	06/16/01*	11weeks after end of fourth quarter
Complete annual public meetings	07/21/01	16weeks after end of fourth quarter
** Approximate date - actual date may vary*		

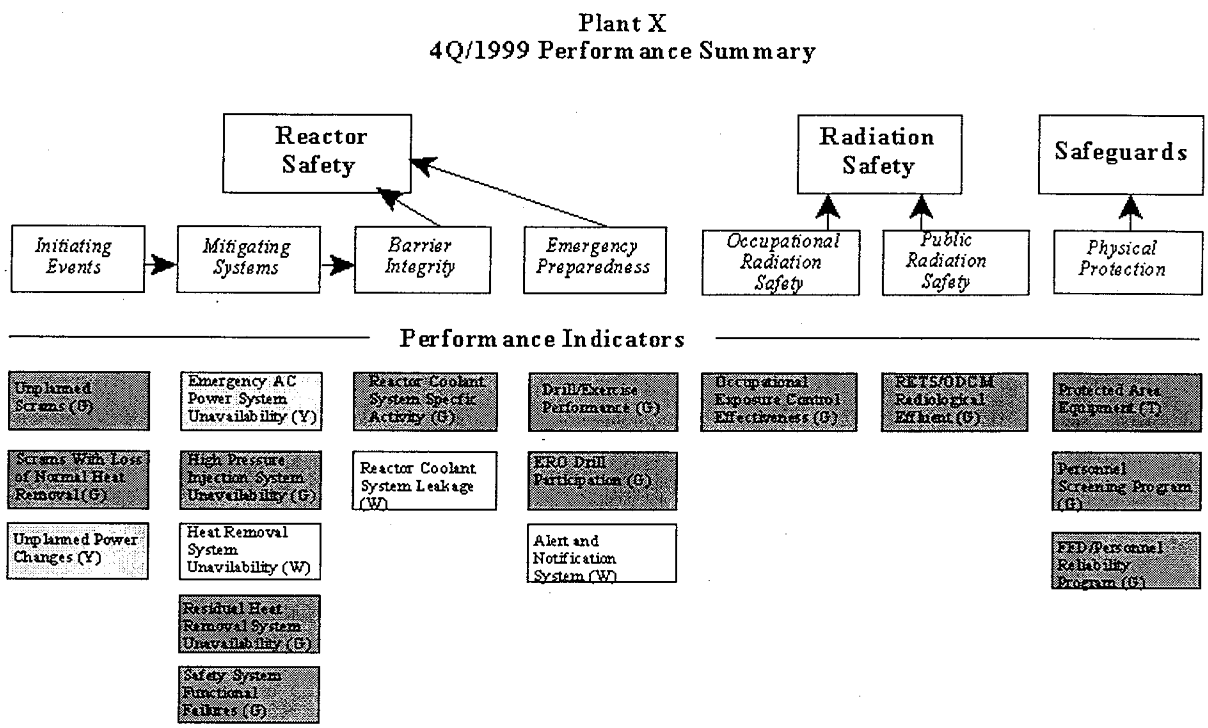
Exhibit 5 - Action Matrix

		Licensee Response Column	**Regulatory Response Column**	**Degraded Cornerstone Column**	**Multiple/Repetitive Degraded Cornerstone Column**	**Unacceptable Performance Column**
RESULTS		All Assessment Inputs (Performance Indicators (PIs) and Inspection	One or Two White Inputs (in different cornerstones) in a Strategic Performance Area;	One Degraded Cornerstone (2 White Inputs or 1 Yellow Input) or any	Repetitive Degraded Cornerstone, Multiple Degraded Cornerstones, Multiple Yellow Inputs, or 1 Red	Overall Unacceptable Performance; Plants Not Permitted to Operate Within this

		Findings) Green; Cornerstone Objectives Fully Met	Cornerstone Objectives Fully Met	3 White Inputs in a Strategic Performance Area; Cornerstone Objectives Met with Minimal Reduction in Safety Margin	Input[1]; Cornerstone Objectives Met with Longstanding Issues or Significant Reduction in Safety Margin	Band, Unacceptable Margin to Safety
RESPONSE	**Regulatory Performance Meeting**	None	Branch Chief (BC) or Division Director (DD) Meet with Licensee	DD or Regional Administrator (RA) Meet with Licensee	RA (or EDO) Meet with Senior Licensee Management	Commission meeting with Senior Licensee Management
	Licensee Action	Licensee Corrective Action	Licensee Corrective Action with NRC Oversight	Licensee Self Assessment with NRC Oversight	Licensee Performance Improvement Plan with NRC Oversight	
	NRC Inspection	Risk-Informed Baseline Inspection Program	Baseline and supplemental inspection procedure 95001	Baseline and supplemental inspection procedure 95002	Baseline and supplemental inspection procedure 95003	
	Regulatory Actions	None	Supplemental inspection only	Supplemental inspection only	-10 CFR 2.204 DFI -10 CFR 50.54(f) Letter - CAL/Order	Order to Modify, Suspend, or Revoke Licensed Activities
COMMUNICATION	**Assessment Reports**	BC or DD review/sign assessment report (w/ inspection plan)	DD review/sign assessment report (w/ inspection plan)	RA review/sign assessment report (w/ inspection plan)	RA review/sign assessment report (w/ inspection plan) Commission Informed	
	Annual Public Meeting	SRI or BC Meet with Licensee	BC or DD Meet with Licensee	RA (or designee) Discuss Performance with Licensee	EDO (or Commission) Discuss Performance with Senior Licensee Management	Commission Meeting with Senior Licensee Management
	INCREASING SAFETY SIGNIFICANCE ---------->					

1. It is expected in a few limited situations that an inspection finding of this significance will be identified that is not indicative of overall licensee performance. The staff will consider treating these inspection findings as exceptions for the purpose of determining appropriate actions.

Exhibit 6



Legend: (G) = Green (T) = Thresholds under development/review
(W) = White (I) = Insufficient data to calculate PI
(Y) = Yellow (N) = Not applicable
(R) = Red (U) = Unique design

Exhibit 6 (Continued)

Plant X
Performance Summary (continued)

Most Significant Inspection Findings

	Initiating Events	*Mitigating Systems*	*Barrier Integrity*	*Emergency Preparedness*	*Occupational Radiation Safety*	*Public Radiation Safety*	*Physical Protection*
4Q/1999	G	G	G	G	G	G	No findings this quarter
3Q/1999	G	G	G	Findings without color designation	G	G	G
2Q/1999	G	Y	Findings without color designation	Findings without color designation	G	G	G
1Q/1999	R	G	W	No findings this quarter	G	G	G

Click here for miscellaneous findings

Additional Inspection & Assessment Information

Assessment Letters:
- 4Q/1999
- 3Q/1999
- 2Q/1999
- 1Q/1999

- Inspection Plans
- Inspection Reports

Exhibit 7

Sample Assessment Follow-Up Letter

Licensee distribution designate
Licensee name/address

SUBJECT: Assessment Follow-Up - **(Plant Name)**

(Use one of the two paragraphs, as appropriate)

(Use the following sentences as appropriate)

1. Our review of **(plant name)** identified that you have crossed the threshold(s) for the **(insert performance indicator(s) threshold crossed)** performance indicator(s). We have identified significant inspection findings in the **(name of cornerstone)** area. Therefore, we plan to conduct additional (supplemental) inspections to better understand the causes contributing to your decline in performance.
2. Our review of **(plant name)** identified that you have crossed the threshold(s) for the **(insert performance indicator(s) threshold crossed)** performance indicator(s). We have identified significant inspection findings in the **(name of cornerstone)** area. However, we do not plan to conduct additional inspections because:

   (State reasons why you will not conduct additional inspections)

This letter is to inform you that we will be planning supplemental inspection at your facility during the month of **(month/year)** to review **(state what area you intend to review).**

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR).

Please contact **(DRP Branch Chief)** at **(telephone number)** with any questions you may have regarding this letter.

(Signed by), Chief
Reactor Projects Branch ____
Division of Reactor Projects

Docket Nos. 50-ABC, 50-XYZ
Licensee Nos. NPF-0, NPF-0
cc:
Normal cc list

Exhibit 8

Sample Mid-Cycle Letter

Licensee distribution designate
Licensee name/address

SUBJECT: Mid-Cycle Review - **(Plant Name)**

On **(date(s))**, the NRC staff reviewed **(plant name)** to integrate performance information and to plan for inspection activities at your facility from **(month/day/year to month/day/year)**. The purpose of this letter is to inform you of our plans for future inspections at your facility so that you will have an opportunity to prepare for these inspections and to inform us of any planned inspections which may conflict with your plant activities.

(Use one of the two paragraphs, as appropriate)

(Use the last two sentences of this paragraph, as appropriate)

1. We have not identified any areas in which you crossed a performance threshold. Therefore, we plan to conduct only baseline inspections at your facility over the next 12 months. However, the significance of (state finding) is still under review via the Significance Determination Process. **(Add additional details, as necessary)**
2. **(Use the following sentences, as appropriate)**

Our review of **(plant name)** identified that you have crossed the threshold(s) for the **(insert performance indicator(s) threshold crossed, color, and risk significance)** performance indicator(s). The staff has identified significant inspection findings in the **(name of cornerstone)** area.

(Additional information on assessment input, as appropriate)

[If these findings have been reviewed by the licensee]

We have conducted additional inspections of your investigation into these findings and we are satisfied with your review and proposed corrective actions.

[If these findings have not been reviewed by the licensee]

Therefore, we will perform additional inspections to review your investigations into these findings and your proposed corrective actions.

or

No additional inspections are planned in **(name of area(s))** because **(basis of decision not to conduct further in this area(s))**

[Add the following paragraph, if appropriate]

Additionally, the staff has identified distinct adverse trends as indicated by substantial cross-cutting issue(s) that have not resulted in performance indicators or inspection findings outside of the licensee response band. **[Provide a qualitative discussion of substantial cross-cutting issues]**

This letter advises you of our planned inspection effort resulting from the **(plant name)** mid-cycle review. Enclosure 1 details the scheduled inspections that will occur from **(month/day/year to month/day/year)**. Enclosure 2 is the plant summary of your performance indicators and inspection findings and enclosure 3 is detailed summary of your performance indicators. Enclosure 4 contains a historical listing of plant issues, referred to as the plant Issues Matrix (PIM), that was used during this review to arrive at our integrated view of your performance. The inspection plan is provided to minimize the resource impact on your staff and to allow for scheduling conflicts and personnel availability to be resolved in advance of inspector arrival onsite. Routine resident inspections are not listed due to their ongoing and continuous nature. The last six months of the inspection plans are tentative and will be revised at the end-of-cycle review meeting.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR). The attached enclosures can also be reviewed at the following NRC website:
http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html

If circumstances arise which cause us to change this inspection plan, we will contact you to discuss the change as soon as possible. Please contact **(DRP Branch Chief)** at **(telephone number)** with any questions you may have regarding this letter or the inspection plan.

(Signed by), Chief
Reactor Projects Branch ____
Division of Reactor Projects

Docket Nos. 50-ABC, 50-XYZ
Licensee Nos. NPF-0, NPF-0

Enclosures:
1. **(Plant name)** Inspection/ Activity Plan
2. **(Plant name)** Plant Summary
3. Detailed summary of **(Plant name)** performance indicators
4. Plant Issues Matrix (PIM)

cc.
Normal cc list
Distribution:
Normal distribution list
plus Chief, NRR/DIPM/IIPB

Exhibit 9

Sample Annual Assessment Letter for Plants in the Licensee Response Column

Report XX-XXXXLicensee distribution designate
Licensee name/address

SUBJECT: Annual Assessment Letter - **(Plant Name)**
On **(date(s))**, the NRC staff completed its end-of-year plant performance assessment of **(plant name).** The end-of-year review for **(plant name)** involved the participation of all technical divisions in evaluating performance indicators (PIs) and inspection results for the period **(month/day/year to month/day/year)**. The purpose of this letter is to inform you of our assessment of your safety performance during this period.

Overall, **(plant name)** operated in manner that preserved public health and safety. **(Plant name)** fully met all cornerstone objectives. While plant performance for the most recent quarter is within the licensee response column of the Action Matrix, there were **(inspection findings and/or performance indicators)** that were outside of the licensee response band during the first three quarters of the assessment cycle. **[Provide a brief summary of inspection findings and performance indicators that were outside of the licensee response band from the first three quarters of the assessment cycle]**

Based on the End-of-Cycle Review results, all performance indicators for the cornerstones were in the licensee response band. Additionally, NRC inspections and licensee self assessments did not identify any findings that were greater than green (very low safety significance) within the cornerstones of safety.

[Add the following paragraph, if appropriate]

Additionally, the staff has identified distinct adverse trends as indicated by substantial cross-cutting issue(s) that have not resulted in performance indicators or inspection findings outside of the licensee response band: **[Provide a qualitative discussion of substantial cross-cutting issues]**

Therefore, we plan to conduct only baseline inspections at your facility over the next 12 months.

This letter advises you of our planned inspection effort resulting from the **(plant name)** end-of-cycle review. Enclosure 1 details the scheduled inspections that will occur from **(month/day/year to month/day/year)**. Enclosure 2 is the plant summary of your performance indicators and inspection findings and enclosure 3 is detailed summary of your performance indicators. Enclosure 4 contains a historical listing of plant issues, referred to as the plant Issues Matrix (PIM), that was used during this review to arrive at our integrated view of your performance. The inspection plan is provided to minimize the resource impact on your staff and to allow for scheduling conflicts and personnel availability to be resolved in advance of inspector arrival onsite. Routine resident inspections are not listed due to their ongoing and continuous nature. The last six months of the inspection plans are tentative and will be revised at the mid-cycle review meeting.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR). The Plant Issues Matrix (PIM) and performance indicators can be reviewed at the following NRC website:

http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html

If circumstances arise which cause us to change this inspection plan, we will contact you to discuss the change as soon as possible. Please contact **(DRP Branch Chief)** at **(telephone number)** with any questions you may have regarding this letter or the inspection plan.

(Signed by), Chief
Projects Branch X
Division of Reactor Projects, Region ___

Docket Nos. 50-ABC, 50-XYZ
Licensee Nos. NPF-0, NPF-0

Enclosures:
1. **(Plant name)** Inspection/ Activity Plan
2. **(Plant name)** Plant Summary
3. Detailed summary of **(Plant name)** performance indicators
4. Plant Issues Matrix (PIM)

cc.
Normal cc list
Distribution:
Normal distribution list
plus Chief, NRR/DIPM/IIPB

Exhibit 10

Sample Annual Assessment Letter for Plants in the Regulatory Response Column

Report XX-XXXXLicensee distribution designate
Licensee name/address

SUBJECT: Annual Assessment Letter - **(Plant Name)**

On **(date(s))**, the NRC staff completed its end-of-year plant performance assessment of **(plant name).** The end-of-year review for **(plant name)** involved the participation of all technical divisions in evaluating performance indicators (PIs) and inspection results for the period **(month/day/year to month/day/year)**. The purpose of this letter is to inform you of our assessment on your safety performance during this period.

Overall, **(plant name)** operated in manner that preserved public health and safety. **(Plant name)** fully met all cornerstone objectives. While plant performance for the most recent quarter is within the regulatory response column of the Action Matrix, there were additional **(inspection findings and/or performance indicators)** that were outside of the licensee response band during the first three quarters of the assessment cycle. **[Provide a brief summary of inspection findings and performance indicators that were outside of the licensee response band from the first three quarters of the assessment cycle]**

[Use either one of the next two sentences, as appropriate, to discuss the PIs]

Based on the End-of-Cycle Review results, all performance indicators for the cornerstones were in the licensee response band.
or
Based on the End-of-Cycle Review results, the performance indicators for the cornerstones were in the licensee response band with the following exceptions:

(Provide PI(s) which crossed the threshold, including color, and risk-significance)

[Use either one of the next two sentences, as appropriate, to discuss NRC inspections]

Additionally, NRC inspections and licensee self assessments did not identify any findings that were greater than green (very low safety significance) in any of the cornerstones.
or
Additionally, NRC inspections identified or confirmed risk significant area(s) in **(name of cornerstone(s)).**

[Provide brief additional information about these findings, as appropriate]

[If these findings have been reviewed by the licensee]

We have conducted additional inspections of your investigation into these events and we are satisfied with your review and proposed corrective actions.

[If these findings have not been reviewed by the licensee]

Therefore, we will perform additional inspections to review your investigations into these events and your proposed corrective actions.
or

No further agency action to these events is warranted because **(state reason(s))**

[Add the following paragraph, if appropriate]

Additionally, the staff has identified distinct adverse trends as indicated by substantial cross-cutting issue(s) that have not resulted in performance indicators or inspection findings outside of the licensee response band. **[Provide a qualitative discussion of substantial cross-cutting issues]**

This letter advises you of our planned inspection effort resulting from the **(plant name)** end-of-cycle review. Enclosure 1 details the scheduled inspections that will occur from **(month/day/year to month/day/year)**. Enclosure 2 is the plant summary of your performance indicators and inspection findings and enclosure 3 is detailed summary of your performance indicators. Enclosure 4 contains a historical listing of plant issues, referred to as the plant Issues Matrix (PIM), that was used during this review to arrive at our integrated view of your performance. The inspection plan is provided to minimize the resource impact on your staff and to allow for scheduling conflicts and personnel availability to be resolved in advance of inspector arrival onsite. Routine resident inspections are not listed due to their ongoing and continuous nature. The last six months of the inspection plans are tentative and will be revised at the mid-cycle review meeting.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR). The Plant Issues Matrix (PIM) and performance indicators can be reviewed at the following NRC website:

http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html

If circumstances arise which cause us to change this inspection plan, we will contact you to discuss the change as soon as possible. Please contact **(DRP Branch Chief)** at **(telephone number)** with any questions you may have regarding this letter or the inspection plan.

(Signed by), Director
Division of Reactor Projects, Region __

Docket Nos. 50-ABC, 50-XYZ
Licensee Nos. NPF-0, NPF-0

Enclosures:
1. **(Plant name)** Inspection/ Activity Plan
2. **(Plant name)** Plant Summary
3. Detailed summary of **(Plant name)** performance indicators
4. Plant Issues Matrix (PIM)

cc.
Normal cc list
Distribution:
Normal distribution list
plus Chief, NRR/DIPM/IIPB

Exhibit 11

Sample Annual Assessment Letter for Plants in the Degraded Cornerstone Column

Report XX-XXXXLicensee distribution designate
Licensee name/address

SUBJECT: Annual Assessment Letter - **(Plant Name)**

On **(date(s))**, the NRC staff completed its end-of-year plant performance assessment of **(plant name).** The end-of-year review for **(plant name)** involved the participation of all technical divisions in evaluating performance indicators (PIs) and inspection results for the period **(month/day/year to month/day/year)**. The purpose of this letter is to inform you of our assessment on your safety performance during this time period.

Overall, **(plant name)** operated in manner that preserved public health and safety. **(Plant name)** met all cornerstone objectives with minimal reduction in the safety margin. However, **(Cornerstone)** was degraded. While plant performance for the most recent quarter is within the degraded cornerstone column of the Action Matrix, there were additional **(inspection findings and/or performance indicators)** that were outside of the licensee response band during the first three quarters of the assessment cycle. **[Provide a brief summary of inspection findings and performance indicators that were outside of the licensee response band from the first three quarters of the assessment cycle]**

[Use either one of the next two sentences, as appropriate, to discuss PIs]

Based on the End-of-Cycle Review results, all performance indicators for the cornerstones were in the licensee response band.
or
Based on the End-of-Cycle Review results, the performance indicators for the cornerstones were in the licensee response band with the following exceptions:

(Provide PIs which crossed the threshold, color, and risk-significance)

[Use either one of the next two sentences, as appropriate, to discuss NRC inspections]

Additionally, NRC inspections and licensee self assessments did not identify any findings that were greater than green (very low safety significance) in any of the cornerstones.
or
Additionally, NRC inspections identified/confirmed risk significant event(s) in **(name of cornerstone(s)).**

[Provide brief additional information about these findings, as appropriate]

[If these findings have been reviewed by the licensee]

We have conducted our own independent inspections of the events which resulted in a degraded cornerstone. Further, we have reviewed your self assessment, conducted with NRC oversight, of the causes contributing to the degraded cornerstone. **(Discuss regional evaluation of licensee self-assessment)**

[If these findings have not been reviewed by the licensee]

Therefore, you should conduct a self assessment into the causes for the degraded cornerstone. Your self assessment efforts should be coordinated with my staff since it will require NRC oversight. Additionally, we will conduct our own independent investigation into the causes for the degraded cornerstone.

[Use either one of the next two sentences, as appropriate]

Because **(cornerstone)** was degraded, this letter is to advise you that we believe a meeting with you would be appropriate. I will be contacting you to arrange for a mutually agreeable time and location for a meeting to discuss your declining performance and your proposed actions to correct these deficiencies.

[Add the following paragraph, if appropriate]

Additionally, the staff has identified distinct adverse trends as indicated by substantial cross-cutting issue(s) that have not resulted in performance indicators or inspection findings outside of the licensee response band. **[Provide a qualitative discussion of substantial cross-cutting issues]**

This letter advises you of our planned inspection effort resulting from the **(plant name)** end-of-cycle review. Enclosure 1 details the scheduled inspections that will occur from **(month/day/year to month/day/year)**. Enclosure 2 is the plant summary of your performance indicators and inspection findings and enclosure 3 is detailed summary of your performance indicators. Enclosure 4 contains a historical listing of plant issues, referred to as the plant Issues Matrix (PIM), that was used during this review to arrive at our integrated view of your performance. The inspection plan is provided to minimize the resource impact on your staff and to allow for scheduling conflicts and personnel availability to be resolved in advance of inspector arrival onsite. Routine resident inspections are not listed due to their ongoing and continuous nature. The last six months of the inspection plans are tentative and will be revised at the mid-cycle review meeting.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR). The Plant Issues Matrix (PIM) and performance indicators can be reviewed at the following NRC website:

http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html

If circumstances arise which cause us to change this inspection plan, we will contact you to discuss the change as soon as possible. Please contact **(DRP Branch Chief)** at **(telephone number)** with any questions you may have regarding this letter or the inspection plan.

(Signed by)
Regional Administrator, Region XX

Docket Nos. 50-ABC, 50-XYZ
Licensee Nos. NPF-0, NPF-0

Enclosures:
1. **(Plant name)** Inspection/ Activity Plan
2. **(Plant name)** Plant Summary
3. Detailed summary of **(Plant name)** performance indicators
4. Plant Issues Matrix (PIM)

cc.
Normal cc list
Distribution:
Normal distribution list
plus Chief, NRR/DIPM/IIPB

Exhibit 12

Sample Annual Assessment Letter for Plants in the Multiple/Repetitive Degraded Cornerstone Column

Report XX-XXXLicensee distribution designate
Licensee name/address

SUBJECT: Annual Assessment Letter - **(Plant Name)**

On **(date(s))**, the NRC staff completed the end-of-year plant performance assessment of **(plant name).** The end-of-year review for **(plant name)** involved the participation of all technical divisions in evaluating performance indicators (PIs) and inspection results for the period **(month/day/year to month/day/year)**. The purpose of this letter is to inform you of our assessment on your safety performance during this time period.

Overall, **(plant name)** operated in manner that preserved public health and safety. **(Plant name)** met all cornerstone objectives with longstanding issues or significant reduction in safety margin. While plant performance for the most recent quarter is within the multiple/repetitive degraded cornerstone column of the Action Matrix, there were additional **(inspection findings and/or performance indicators)** that were outside of the licensee response band during the first three quarters of the assessment cycle. **[Provide a brief summary of inspection findings and performance indicators that were outside of the licensee response band from the first three quarters of the assessment cycle]**

[Use either one of the next two sentences, as appropriate, to discuss PIs]

Based on the End-of Cycle Review results, all performance indicators for the cornerstones were in the licensee response band.
or
Based on the End-of Cycle Review results, the performance indicators for the cornerstones were in the licensee response band with the following exceptions:

(Provide PIs which crossed the threshold, color, and risk-significance)

[Use either one of the next two sentences, as appropriate, to discuss NRC inspections]

Additionally, NRC inspections and licensee self assessments did not identify any findings of greater than

green (very low safety significance) in any of the cornerstones.
or
Additionally, NRC inspections identified/confirmed risk significant findings in **(name of cornerstone(s)).**

[Provide brief additional information about these findings, as appropriate]

Therefore, you should develop a performance improvement plan which will correct the deficiencies which are causing degradation of your cornerstones. Your implementation of the performance improvement plan should be coordinated with my staff since it will require the formation of an NRC Oversight Panel. Additionally, we will be conducting our own independent team investigation into the causes for the degraded cornerstone(s) which will be coordinated through the Oversight Panel.

Because **(cornerstone(s))** was/were degraded, this letter is to advise you that we believe a meeting between the Executive Director for Operations and your senior management would be appropriate. I will be contacting you to arrange for a mutually agreeable time and location for a meeting to discuss your declining performance and your proposed actions to correct these deficiencies.

[Add the following paragraph, if appropriate]

Additionally, the staff has identified distinct adverse trends as indicated by substantial cross-cutting issue(s) that have not resulted in performance indicators or inspection findings outside of the licensee response band. **[Provide a qualitative discussion of substantial cross-cutting issues]**

This letter advises you of our planned inspection effort resulting from the **(plant name)** end-of-cycle review. Enclosure 1 details the scheduled inspections that will occur from **(month/day/year to month/day/year)**. Enclosure 2 is the plant summary of your performance indicators and inspection findings and enclosure 3 is detailed summary of your performance indicators. Enclosure 4 contains a historical listing of plant issues, referred to as the plant Issues Matrix (PIM), that was used during this review to arrive at our integrated view of your performance. The inspection plan is provided to minimize the resource impact on your staff and to allow for scheduling conflicts and personnel availability to be resolved in advance of inspector arrival onsite. Routine resident inspections are not listed due to their ongoing and continuous nature. The last six months of the inspection plans are tentative and will be revised at the mid-cycle review meeting.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR). The Plant Issues Matrix (PIM) and performance indicators can be reviewed at the following NRC website:

http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html

If circumstances arise which cause us to change this inspection plan, we will contact you to discuss the change as soon as possible. Please contact **(DRP Branch Chief)** at **(telephone number)** with any questions you may have regarding this letter or the inspection plan.

(Signed by)
Regional Administrator, Region XX

Docket Nos. 50-ABC, 50-XYZ

Licensee Nos. NPF-0, NPF-0

Enclosures:
1. **(Plant name)** Inspection/ Activity Plan
2. **(Plant name)** Plant Summary
3. Detailed summary of **(Plant name)** performance indicators
4. Plant Issues Matrix (PIM)

Normal cc list
Distribution:
Normal distribution list
plus Chief, NRR/DIPM/IIPB

ATTACHMENT 6

Issue Date: 04/21/00

Attachment 0609.02

Initial Assessment of Inspection Observations for SDP Entry

Issues that have an insignificant effect on plant risk or otherwise do not merit documentation in an NRC inspection report are classified as minor issues. Classifying issues as minor requires inspector judgement. The guidance in IMC 0610 Appendix H is the most recent information and best examples of what constitutes minor issues. However, in general the inspector can use the questions listed below as a filter to determine if an issue can be considered minor.

Minor Issues	**Group 1 Questions**
Does the issue have an actual or credible impact on safety?	
Does the issue suggest a programmatic problem that has a credible potential to impact safety and is more than an isolated case?	
Could the issue be viewed as a precursor to a significant event?	
If left uncorrected would the same issue become a more significant safety concern ?	
Are there any associated circumstances that add regulatory or safety concerns, (eg. apparent willfulness, licensee refusal to comply)?	
Does the issue relate solely to NRC limits and not licensee administrative limits?	
Does the issue relate to performance indicators and causes a threshold to be exceeded?	

If the answer to all the above questions is "No", the issue may be considered minor. The issue should be discussed with the licensee but not documented in the report.

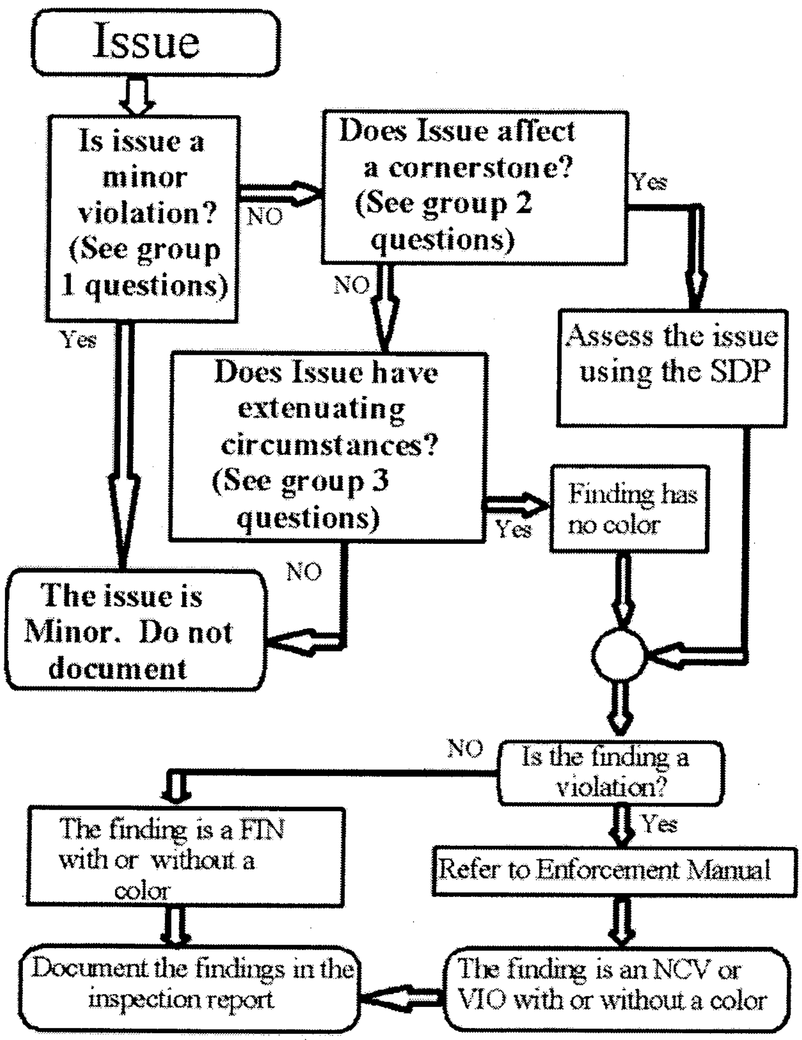
The group 2 questions should be used to determine whether an issue affects a Cornerstone. If the answer to any single question is "yes", the issue should be analyzed by the SDP process and documented in the inspection report. If the answers to all group 2 (Cornerstone questions) are "no" Then the inspector should determine whether there are extenuating circumstances by asking the Group 3 questions.

Cornerstones	Group 2 Questions
Reactor Safety - Initiating Events, Mitigating Systems, & Barrier Integrity	
Could it cause or increase the frequency of an initiating event?	
Could it affect the operability, availability, reliability or function of a system or train in a mitigating system?	
Could it affect the integrity of fuel cladding, the reactor coolant system, and/or reactor containment?	
Could it involve degraded conditions that concurrently influence any mitigation equipment and/or initiating event?	
Reactor Safety - Emergency Planning	
Does it involve a failure to meet or implement a planning standard (10CFR50.47(b) and Appendix E to Part 50) or other regulatory requirement?	
Radiation Safety - Occupational	
For ALARA issues: (a) Does the actual job dose exceed the projected dose by >50%, AND (b) is the 3 year rolling average collective dose exceed 135 person-rem/unit for a PWR or 240 person-rem/unit for a BWR, AND (c) is the actual job dose > 5 person-rem?	
Does it involve a failure of one or more radiation barriers that result in, or could result in, a significant unintended or unplanned dose ?	
Radiation Safety - Public	
Does it involve an occurrence in the licensee's radiological*effluent monitoring* program that is contrary to NRC regulations or the licensee's TS, ODCM, or procedures?	
Does it involve an occurrence in the licensee's radiological*environmental monitoring* program that is contrary to NRC regulations or the licensee's TS, ODCM, or procedures?	
Does it involve an occurrence in the licensee's radioactive*material control* program that is contrary to NRC regulations or the licensee's procedures?	
Does it involve an occurrence in the licensee's radioactive material transportation program that is contrary to NRC or DOT regulations or licensee procedures?	
Physical Protection	
Does it involve a nonconformance with safeguards requirements?	
Fire Protection	
Does it involve impairment or degradation of a fire protection feature or defense-in-depth?	

If the answer to any question is "Yes", the issue affects a cornerstone and should be analyzed by the associated SDP.

Extenuating Circumstances	**Group 3 Questions**
Does the issue involve willfulness, including discrimination?	
Does the issue have potential for impacting the NRC's ability to perform its regulatory function?	
Is documenting this issue necessary to close an open item, licensee event report or allegation?	
Does the associated technical information relate directly to an issue of agency-wide concern, i.e. a generic safety issue?	
Does the issue provide substantive information regarding cross cutting issues?	
Is the finding a violation?	

If all the answers to the above questions are "No", the issue does not have extenuating circumstances and would not normally be documented.



Issue Date: 10/06/00

NRC INSPECTION MANUAL IIPB

MANUAL CHAPTER 0610*

Power Reactor Inspection Reports

Table of Contents

Inspection Reports

0610-01 Purpose

To give guidance on content, format, and style for reports of power reactor inspections.

0610-02 Objectives

To ensure that inspection reports:

02.01 Clearly communicate significant inspection results to licensees, NRC staff, and the public.

02.02 Provide a basis for significance determination and enforcement action.

02.03 Present information associated with significant inspection findings in a manner that will be useful to NRC management in developing longer-term, broad assessments of licensee performance.

0610-03 Definitions

The following terms are applicable to the enforcement program.

Apparent violation. A potential noncompliance with a regulatory requirement (regardless of possible significance or severity level) that has not yet been formally dispositioned by the NRC.(All inspector identified violations greater than the level of an NCV are initially apparent violations).

Closed Item. A matter previously reported as a noncompliance, an inspection finding, a licensee event report, or an unresolved item, that the inspector concludes has been satisfactorily addressed based on information obtained during the current inspection.

Credible. A scenario offering reasonable grounds for being realistic (given a set of existing conditions postulating a scenario with no more than one "if").

Cross-Cutting Issues. Cross-cutting issues are those concerns related to the areas of human performance, problem identification and resolution, and safety-conscious work environment which have the potential to affect multiple cornerstones.

Deficiency. (Applies to emergency preparedness) A demonstrated level of performance (e.g., in a drill) that could have detracted from effective implementation of the emergency plan in the event of an actual emergency.

Deviation. A licensee's failure to satisfy a written commitment, such as a commitment to conform to the provisions of applicable codes, standards, guides, or accepted industry practices when the commitment, code, standard, guide, or practice involved has not been made a requirement by the Commission.

Escalated Enforcement Action. A notice of violation or civil penalty for any Severity Level I, II, or III violation (or problem); a notice of violation associated with an inspection finding that the significance determination process characterizes as having low to moderate, or greater safety significance; or any order based upon a violation.

Finding. An issue with some significance that has been placed in context and determined either to be of sufficient significance to warrant more detailed analysis using the SDP or to have extenuating circumstances warranting its documentation in an inspection report. To be a finding, it must pass through the threshold screening process described in Appendix B, "Threshold for Documentation", in this Manual Chapter. Findings may or may not be related to regulatory requirements.

Green Finding. A finding of very low safety significance.

Independent Item. An item used to track information that does not originate in or is typically documented in an inspection report but may be used to assess plant performance such as an Office of Investigation harassment and intimidation case.

Integrated Inspection Reports. A reactor inspection report that combines inputs from several inspections (resident, regional, etc.) conducted within a specific period.

Issue. A well-defined observation or collection of observations which are of concern and may or may not result in a finding.

Minor Violation/ Finding. A violation or finding that is less significant than either a Severity Level IV violation or less significant than a finding which the significance determination process characterizes as Green (very low safety significance). Although minor violations must be corrected, they are not usually described in inspection reports.

Non-Cited Violation (NCV). A method for dispositioning a Severity Level IV violation or a violation associated with a finding that the significance determination process characterizes as Green (very low safety significance). Provided applicable criteria in the Enforcement Policy are met, such issues are documented as violations, but are not cited in notices of violation which normally require written responses from licensees.

Noncompliance. A violation (regardless of whether they are cited or not), nonconformance, or deviation.

Nonconformance. A vendor's or certificate holder's failure to meet contract requirements related to NRC activities (e.g., 10 CFR Part 50, Appendix B, Part 71, or Part 72) where the NRC has not placed requirements directly on the vendor or certificate holder.

Notice of Violation (NOV). A formal written citation in accordance with 10 CFR 2.201 that sets forth one or more violations of a legally binding regulatory requirement.

Observation. A fact; any detail noted during an inspection.

Open Item. A matter that requires further inspection or evaluation. The reason for requiring further inspection or evaluation may be that the matter has been identified as an unresolved item.

Potentially Generic Issue. An inspection finding that may have implications for other licensees, certificate holders, and vendors whose facilities or activities are of the same or similar manufacture or style.

Red finding. A finding of high safety significance.

Significance. The quality of being important: As used in this Inspection Manual Chapter (IMC), it involves the consideration of (1) actual safety consequences; (2) potential safety consequences, including the consideration of risk information; (3) potential for impacting the NRC's ability to perform its regulatory function; and (4) any willful aspects of the violation.

Significance Determination. The characterization of the significance of an inspection finding using the significance determination process (SDP) outcome color scheme to identify the level of safety significance (i.e., Green, White, Yellow, Red).

Significance Determination Process (SDP). The process used to determine the risk or safety significance of pertinent inspection findings within the reactor oversight process.

Significant. Having or likely to have influence or effect. For example, a White issue still under review is an apparent significant issue with low to moderate safety significance.

Substantive. Involving matters of major or practical importance; considerable in amount or numbers. In this manual chapter substantive information must be placed in context relative to the inspection scope and the potential or actual safety significance.

Unresolved Item. A matter about which more information is required to determine whether the issue in question is an acceptable item, a deviation, or a violation, or for which the significance has not yet been determined: such a matter may require additional information from the licensee or cannot be resolved until additional guidance or information is obtained such as through a task interface agreement (TIA), or other policy determinations.

Violation. The failure to comply with a legally binding regulatory requirement, such as a statute, regulation, order, license condition, or technical specification.

Weakness. (Applies to emergency preparedness.) A demonstrated level of performance (e.g., in a drill) that could have precluded effective implementation of the emergency plan in the event of an actual emergency.

Willfulness. An attitude toward non-compliance with requirements that ranges from careless disregard to a deliberate intent to violate or to falsify.

White Finding. A finding of low to moderate safety significance.

Yellow Finding. A finding of substantial safety significance.

0610-04 Responsibilities

All NRC inspectors are required to prepare inspection reports in accordance with the guidance provided in this inspection manual chapter. General and specific responsibilities are listed below.

04.01 General Responsibilities for Power Reactor Inspections. Each inspection of a reactor facility shall be documented in a report consisting of a cover letter, a cover page, a summary of findings, and inspection details.

04.02 Report Writing

a. Inspectors have the primary responsibility for ensuring that inspection findings are accurately reported, and that referenced material is correctly characterized. Advice, subjective opinions, and recommendations are not to be included in inspection reports.
b. Inspectors are responsible for ensuring that the content of the report does not conflict with the information presented at the exit meeting. When the report provides information that differs significantly from that presented at the exit meeting, the inspector (or the report reviewer) should discuss those differences with the licensee before the report is issued.
c. Report writers and reviewers should ensure that inspection reports follow the general format given in this chapter and displayed in the enclosed sample report (see Exhibits 1 and 2).
d. For inspections conducted by regional and resident inspectors, the report number is to be identified in the following form:

   Docket No./Year - [sequential number of the report in that year] (e.g., 50-363/00-01)

   For inspections conducted by NRR, or other headquarters offices, the report number is to be identified in the following form:

   Docket No./Year - 2 [sequential number of the report in that year] (e.g., 50-250/00-201)

04.03 Report Review and Concurrence

a. Before issuance, each inspection report shall, as a minimum, be reviewed by a member of NRC management familiar with NRC requirements in the area inspected.

b. The report reviewer (i.e., the member of management referred to above) shall ensure that inspector findings are consistent with NRC policies and requirements and that enforcement-related issues are addressed in accordance with the NRC Enforcement Policy and the NRC Enforcement Manual.

c. The report reviewer shall ensure that assessments made in the inspection report are in accordance with the SDP.

d. Regional administrators and office directors shall establish internal procedures to provide a record of inspectors' and reviewers' concurrences. The procedures should address how to ensure continued inspector concurrence when substantive changes are made to the report as originally submitted, and how to treat disagreements that occur during the review process. As a minimum, substantial changes shall be discussed with the inspector or inspectors involved to ensure continued concurrence, and disagreements that cannot be adequately resolved shall be documented.

NOTE: The record of inspector and reviewer concurrence is maintained by the issuing office. This concurrence record is not included in the distributed version of the report.

04.04 Report Issuance. For regional inspection reports, the applicable division director or designated branch chief is responsible for the report content, tone, and overall regulatory focus. For integrated reports, the Director, Division of Reactor Projects (DRP), or the designated branch chief is responsible for issuing the report.

04.05 Report Timeliness

a. General Timeliness Guidance. Inspection reports should be issued no later than 30 calendar days after inspection completion (45 calendar days for integrated reports and major team inspections).

NOTE: For non-resident conducted inspections, inspection completion is normally defined as the day of the exit meeting. For resident inspector and integrated inspection reports, inspection completion is normally defined as the last day covered by the inspection report.

b. Reports Preceding Escalated Enforcement Actions. Timeliness goals should be accelerated for inspection reports covering potential escalated enforcement actions.

c. Expedited Reports for Significant Safety Issues. Whenever an inspector identifies issues of greater safety significance (i.e., White or higher) or a significant or immediate public health and safety concern, an expedited inspection report should be considered that is limited in scope to the specific issue. IMC 0609 allows for issues of significance to be documented on an expedited basis.

0610-05 Guidance-Inspection Report

This section relates primarily to the details contained in the inspection report. Refer to Exhibit 2 as a general example (Note: Report details will be added to Exhibit 2 in a future revision to this IMC after experience is gained).

Although this guidance applies to all power reactor inspections, additional guidance for reports documenting supplemental inspections is found in Appendix C and in Appendix D for guidance on inspection reports associated with IP 71152, "Problem Identification and Resolution".

Whenever possible, the Details section of routine and integrated NRC inspection reports should conform to the standard format described in this section and illustrated in the attached Exhibit 1. This standardization in format significantly enhances consistency, readability, and information retrieval, which in turn increases efficiency and improves the ability to integrate inspection results. Exceptions include major team inspection reports, augmented inspection team (AIT) and special inspection reports, supplemental inspections, and other cases where the specifically directed focus of the inspection does not easily fit into the baseline inspection program and subtopics given in the standardized report outline. In these cases and in the cover letters of inspection reports where a standard format is not readily applied, the most important subject should be identified first, followed by a discussion of major topics identified in descending order of significance.

Guidance and cover letter format for enforcement issues vary. Guidance and sample cover letters are found in the Enforcement Manual, Appendix B, "Standard Formats for Enforcement Packages." The following guidelines apply to what should be documented in the cover letter, the summary of findings, and the details of the report.

(1) Findings and violations whose significance is known are to be discussed in the report details, summary of findings, and in the cover letter. The significance is either a color as defined by the SDP evaluation, no color or a severity level for non-SDP violations. If the finding is other than Green, the significance evaluation paragraph should state that "the significance of this item is preliminarily (White or Yellow or Red).

(2) Findings (including violations) whose final significance is not yet determined but is known to be at least Green, are considered unresolved items and should be discussed in the report details, summary of findings, and in the cover letter. The significance is entered in the summary of findings as "TBD" as a lead in color.

(3) Findings whose significance is known from the SDP to be at least Green but the compliance aspect has not yet been determined are considered unresolved items and should be discussed in the report details, summary of findings, and in the cover letter. The significance is the SDP evaluation color or TBD. Additional action may be required to (1) determine whether a non-compliance exist, (2) to update the Plant Issues Matrix (PIM) and, (3) take other associated actions for findings greater than Green.

(4) Unresolved items whose significance has not been evaluated by the SDP should be documented in the report, but not documented in the summary of findings or mentioned in the cover letter. These items are identified as unresolved items (URIs) in the report.

(5) Independent Items are used to track items/information from sources other than inspections (e.g., final SDP letters, OI discrimination letters). They should be documented under 4OA5, "Other."

05.01 Cover Letter. Three example cover letters for reports with (1) no findings,(2) White findings, and (3) Green findings with NRC identified NCVs are provided with the example routine report.

Inspection reports are transmitted using a cover letter from the applicable NRC official (branch chief, division director, or regional administrator) to the designated licensee executive. Cover letter content varies somewhat depending on whether or not the inspection identified noncompliances. In general, however, every cover letter has the same basic structure.

a. Addresses, Date, and Salutation. At the top of the first page, the cover letter begins with the NRC seal and address, followed by the date on which the report cover letter is signed and the report issued.

For cover letters transmitting reports with issues assigned an enforcement action (EA) number, the EA number should be placed in the upper left-hand corner above the principal addressee's name.

The name and title of the principal addressee are placed at least four lines below the letterhead, followed by the licensee's name and address. Note that the salutation is placed after the subject line.

b. Subject Line. The subject line of the letter should state the plant name (e.g.,"DIRJAC Generating Station- NRC INSPECTION REPORT") followed by the report number. The words "NOTICE OF VIOLATION" (or "NOTICE OF DEVIATION," etc.) should be included if such a notice is accompanying the inspection report.

c. Introductory Paragraphs. The first two paragraphs of the letter should give a brief introduction.

d. Body of the Letter. In keeping with the "Plain English Initiatives" which implements the requirements of SECY-99-070 "Implementation Plan for the Public Communications Initiative (DSI-14), the most important topics should be discussed first. White findings or above, for which the issuance of a notice of violation is being considered, should be briefly discussed in the order of their significance. The appropriate wording for issues that are also violations of requirements is included in the Enforcement Manual (under Guidance Documents). If Non-Cited Violations were identified, the report should state that these items were not cited due to their very low safety significance and because they have been entered into the licensee's corrective action system. If Green findings, other than violations, were identified, including unresolved items which have been evaluated by the SDP, the report should state: " There were [the number] findings of very low safety significance (Green) identified in the report;" without further elaboration. If there are no findings in the inspection report, the final statement in this paragraph should state: "Based on the results of this inspection no findings of significance were identified."

e. Closing. The final paragraph consists of standard legal language that varies based on whether or not enforcement action is involved, (See example cover letters in Exhibit 2).

The signature of the appropriate NRC official is followed by the docket number(s), license number(s), and lists of enclosures and distribution.

05.02 Cover Page. The report cover page gives a quick-glance summary of information about the inspection (see Exhibit 2). It contains the dates of inspection, the report number, the names and titles of participating inspectors, and the name and title of the approving NRC manager.

05.03 Summary of Findings. The summary should be informative but concise. The inspection report summary is an overview of the significant inspection findings. It also provides the text for entries to the PIM and Agency Document Access and Management System (ADAMS). The first paragraph is an input into the NRC ADAMS template to improve public access to inspection reports.

a. ADAMS Template. The first paragraph of the summary of findings is used in the title value field of the ADAMS template NRC-002 as a report summary. The paragraph must be cryptic, without the use of extraneous words or articles, and include in the following order: (1) the inspection report number (note the format in example EX2); (2) the dates of the inspection; (3) the name of the utility; (4) the name of the site; and (5) the titles only of the inspection procedures or attachments in which findings were identified (e.g, equipment alignment, fire protection, operability evaluations.) If no findings were identified, then the general inspection area should be listed (e.g, radiation specialist report, or resident inspector report, or environmental report.) This information must be a concise, single paragraph because the field in the ADAMS template is limited to 256 characters.

For non-routine inspections, the same format should be followed for identifying the report number, utility and unit names, and dates of inspection. These are followed by the title of the inspection and a list of findings. (See Appendix C and D for examples).

b. Summary Paragraph. A paragraph following the ADAMS template paragraph describes who conducted the inspection (i.e., resident or specialist inspectors), the number of findings and violations, and a statement that the significance of most (or all) findings described in the report was determined using the significance determination process.

c. Findings. The body of the summary of findings should be compiled by reviewing each report section and writing a summary of each finding, noncompliance, unresolved item, or apparent violation. All findings except licensee identified NCVs or green findings and those that could result in an acceptable conclusion should be included in the summary of findings. Specific requirements violated should also be cited.

   Each finding's summary begins with the significance color (using TBD for those findings whose significance has not yet been determined) or No Color for non-SDP findings. This indication of safety significance is followed by one paragraph that briefly describes the finding, followed by a second paragraph that briefly describes the regulatory nexus or safety evaluation of the finding. If the finding has no color, the second paragraph should describe why the finding is considered to be significant.

   The findings summaries are listed by cornerstones in the order specified in Exhibit 1. Cross-cutting issues are documented as described in Section 06.02. SDP analyzed findings that have a crosscutting element as a causal factor are summarized under the appropriate cornerstone heading. Significant trends in cross cutting areas (based on multiple findings) that are determined to be separate findings are summarized under Section 4OA4.

   Inspectors should ensure that the text of the summaries is consistent with the details and that each summary ends with a reference to the section of the report details where the finding is discussed.

d. Plant Issues Matrix (PIM). The PIM is a consolidated listing of plant issues (i.e., inspection findings) in the Reactor Program System (RPS) that are used by the NRC to assess plant performance. All the entries in the summary of findings are transferred directly to the RPS and designated for the PIM, except for the color of the finding and the reference to the report details paragraph. Although the RPS and PIM are not directly a part of the inspection report, instructions are included here to help inspectors identify during the inspection the information required for the PIM.

   The PIM shall be updated within 14 days after the date of the report and shall include the following information: type, title, cornerstone, significance determination, date, who identified the finding (NRC or licensee), item description and significance description, and source document (normally expected to be the inspection report number). Data will be entered into the PIM via the Reactor Program System/Item Reporting (RPS/IR) module. Detailed guidance on entering and updating PIM entries using RPS/IR will be included in a future IMC titled "Information Technology Support."

   The information from the summary of findings and licensee identified NCVs from section 4OA7 as appropriate shall be transferred to the PIM as written with only minor editorial changes. PIM entries may be changed; however, only information contained in the body of the report shall be used. Care should be taken to ensure that new or undocketed information is not inadvertently introduced into the PIM. Any changes to the facts stated in a PIM entry shall be included within brackets [ ] to clearly show the editing. If the meaning of a PIM entry is confusing after the inspection report is issued, the PIM may be edited to clarify the finding and to improve the reader's understanding of the issue. Brackets are not necessary for edits that only clarify a PIM entry.

   Issues whose significance is known are entered in the PIM with the applicable type code of finding (FIN), violation (VIO), or Non-Cited Violation(NCV). The color of the finding (for SDP issues) or the severity level of the violation (for non-SDP issues) is entered in the significance field. The appropriate cornerstone is designated.

   Issues initially categorized as having a potential safety significance of greater than very low significance (i.e., potentially other than Green), but whose significance has not yet been made final, should be categorized in the RPS significance field as TBD. The type code should be FIN (or AV for apparent violation, if applicable), and the appropriate cornerstone entered into the cornerstone field. After the risk is finally determined by the SDP oversight panel following a regulatory conference (if held) and a letter with that determination is sent to the licensee, the RPS significance field is changed from TBD to the appropriate color. Similarly, after a final enforcement decision is made for issues initially categorized as apparent violations, the type code is changed from AV to VIO. In both cases, text should be added to the original PIM entry that describes the final SDP conclusion and enforcement actions with references to the docketed correspondence.

Unresolved items (URIs) There are various types of URIs, however each is documented in an inspection report, and assigned a tracking number. See Section 0610-05 (2) (3) and (4). If either the significance is known or the compliance aspect is known, they are also entered into RPS. For those that have not been evaluated by the SDP the significance field in RPS for the URI is TBD. the item may be marked for entry into the PIM at a later date, it is not considered in the assessment process. The PIM entry should be made once the issue is resolved and the resolution is documented in an inspection report or other docketed correspondence.

Independent items are used to track items or information from sources other than inspection reports, such as final SDP letters and OI discrimination letters, or to track items given to another organization to follow up. To enter independent items, they must be referenced in an inspection report and entered into RPS through RPS/IR. They are documented in Section 4OA5, "Other," of the next resident inspection report. For SDP issues, the original PIM entry is updated to reflect the disposition described in the final SDP letter. The text added to the PIM entry describes the final SDP conclusion and any enforcement actions, and references the docketed final SDP letter. The RPS significance field for the PIM entry is changed from TBD to G/W/Y/R, as appropriate, and the RPS type code is changed to the appropriate type if applicable (for example from AV to VIO).

Issues related to problem identification and resolution (PI&R) that are identified during routine baseline inspections and documented in inspection reports are in the PIM as part of the RPS entry for the associated inspection finding. Conclusions made on PI&R effectiveness resulting from these routine inspections are not included in the PIM, except to the extent they are associated with an individual inspection finding or contribute to a significant cross-cutting issue as described in Section 06.02 of this manual chapter. However, a summary conclusion regarding the effectiveness of the PI&R program resulting from the annual PI&R inspection (IP 71152) is entered into the PIM with Miscellaneous in the cornerstone field and N/A in the significance field.

Issues from verifying performance indicators (PIs) are entered in the PIM only if correcting the data causes or would cause the PI to cross a threshold. They are documented in the PIM under Miscellaneous in the cornerstone field, VIO or NCV in the type field, and the severity level of the violation in the significance field. Each PI verification issue is a separate PIM entry. Neutral or positive PI verification issues, or issues where the correction of the PI data does not cause the PI to cross a threshold, are not designated for the PIM.

A paragraph summarizing the results of a supplemental inspection of a White, Yellow or Red inspection finding is added to the PIM entry for the original inspection finding. A paragraph summarizing the results of a supplemental inspection performed to address a White, Yellow or Red performance indicator is designated for the PIM under the cornerstone associated with the performance indicator. In general, no color will be assigned to either of these PIM entries, unless a new SDP characterized issue was found during the supplemental inspection.

05.04 Table of Contents. For reports which are considered complicated or are of significant length (i.e., the Report Details section is more than 20 pages long), the writer should include a table of contents as an aid to clarity.

05.05 Report Arrangement. The standardized report outline is provided as Exhibit 1 to this manual chapter. Inspection reports may begin with a Summary of Plant Status section. The section briefly describes pertinent operational issues such as any plant shutdowns or significant changes in power. For specialist inspections, this summary is not needed (e.g., plant operating status may or may not be relevant to a safeguards or emergency preparedness inspection). The report details should be topically arranged in accordance with the standardized report outline. This does not mean that each outline topic should be covered in each report. To the extent that inspection is performed in a particular area (e.g., inspection of "gaseous and liquid effluents"), the resulting findings should be placed in the corresponding standard section of the report (e.g., in 2PS1 of the standardized outline in Exhibit 1).

NOTE: For events the discussion of the entire event should be included under 4OA3 Event Follow-up. However, situations may arise where circumstances surrounding an event or related issues are documented in an another cornerstone area. In this case the event description should be referenced under section 4OA3. For example:

"4OA3 Event Follow-up

.1 Section 2PS1 describes the circumstances and licensee actions regarding a release of gaseous effluents which exceeded 10 CFR Part 20 limits."

05.06 Report Details. The overall organization of each report section should follow the same basic progression of inspectable area, optional title, scope, and findings, as will be shown in the attached sample report (Exhibit 2).

a. Inspection Scope. This section includes a list of items or activities inspected in sufficient detail to inform the reader of what was inspected and what criteria were used to determine the acceptability of what was inspected. The scope should be derived from the inspection objectives and requirements sections from the applicable inspection procedure. Generally, inspection criteria include requirements, codes, industry standards and licensee administrative procedures or drawings (or in some cases the inspection procedure).

   In cases where there are "no significant findings," additional detail should be provided to inform the reader of the methods of inspection as well as objectives and criteria used. Typical methods are a walk down, an in office review, observation of test from the control room, or participation in an exercise.

b. Findings. This portion of each inspectable area of the report is used to document the inspection results. Within each inspectable area the report should discuss the most important finding first. If the inspector identifies no findings during an inspection (other than minor issues), then in the corresponding section of the report, under Findings the inspector should enter "No findings of significance were identified." Minor issues, which may have been identified and discussed with the licensee, and licensee identified Green findings are normally not documented except as noted in 06.03.b.

   When findings are identified, the first sentence or two of this section provides the results of the inspection in the area. This introductory sentence is briefer than the summary of findings and does not need to stand alone because the discussion that will follow will provide the supporting details.

   The next paragraph should provide the description of the finding. The description may consist of several paragraphs depending on the significance and complexity of the finding. This section is to be followed by a significance evaluation paragraph that describes the logic for entering the SDP. That is, it answers the pertinent group 1, 2, or 3 "thresholds for documentation" questions found in Appendix B of this manual chapter. For example:

   "This finding, if left uncorrected, would become a more significant concern and could cause an increase in the frequency of an initiating event because...."

The example above answers the group 1 question that helped the inspector determine that the finding was more than minor, and the group 2 question that helped the inspector determine that the issue affected a cornerstone. If applicable, a group 3 question would be answered to help determine if the finding had extenuating circumstances. This paragraph should also discuss the results of the significance determination.

The concluding paragraph states any associated enforcement actions and references the requirements violated. The paragraph gives the licensee's corrective action program number for the issue to aid the NRC in locating the licensee's corrective action during a later inspection. The enforcement action must be consistent with the preceding significance determination. For example:

> "This finding did have a credible impact on safety; however, since only the initiating event cornerstone is affected and associated assumptions have no other impact than slightly increasing the likelihood of an uncomplicated reactor trip, the finding is considered to be of very low safety significance (Green). The inspectors also determined that, at the time of the event, procedure DOP 512 was not appropriate to the circumstances, constituting a violation of 10 CFR Part 50, Appendix B, Criterion V, "Procedures." However because of the very low safety significance of the item and because the licensee has included this item in their corrective action program (CAP ref. Xxx-xx-2000), this procedure violation is being treated as a Non-Cited Violation (NCV XXX/99007-02)."

For White, Yellow or Red findings the report details present the assumptions supporting an SDP determination, including pertinent issues such as duration, mitigation capabilities, accident scenarios, and worst-case safety significance. Clearly indicate in discussions of accident scenarios and worst-case safety significance if the condition actually occurred or could have credibly occurred. The following guidance applies to providing the appropriate level of detail for documenting complex Green findings or White, Yellow or Red findings.

1. The degree of actual or potential safety consequence associated with a finding should be a primary consideration in determining the level of appropriate detail. Items of potential significance (issues assessed using the reactor SDP phase 2 or similar issues) merit more discussion.
2. Findings likely to have generic concerns should include details such as manufacturer's model number for components, specifications, and other technical data that identifies the item of concern.
3. Findings related to cross-cutting areas must be related to other previously identified or contemporaneous findings that have been analyzed using the SDP. Cross-cutting issues should be discussed in sufficient detail to communicate the nexus or causal relationship to the other findings.
4. If an inspector determines that a finding has added significance based on risk, that perspective should be explained. For example, if the inspector finds that two components with reliability problems are related by a dominant event sequence, that relationship should be explained.
5. Positive issues should not be documented. However, when describing all the information needed to properly perform an SDP evaluation, those licensee actions that mitigate a potential problem should be supported by the appropriate description of positive licensee performance that influenced the significance of the finding.
6. When documenting an unresolved item, the issue description should provide enough background information that a different inspector, using that information, would be able to perform the follow-up inspection.
7. If an issue found during an inspection is to be referred to OI, the inspection report should not lead a reader to conclude or infer that an OI investigation is possible. For issues referred to OI, the report should contain only relevant factual information collected during the inspection. The referral to OI is made by correspondence separate from the inspection report and includes any additional information needed to support the referral. Any reports containing material that may be related to an on-going investigation should be reviewed by OI before it is issued. An internal record of OI concurrence

according to Section 04.03(d) is retained.

Uncomplicated Green findings should be succinctly described in less than a page. Complex Green issues should be described in no more than 2 pages. More significant findings may need more documentation because of their complexity and significance.

05.07 Exit Meeting Summary. The final section of each reactor inspection report briefly summarizes the exit meeting. It identifies the licensee manager who attended the meeting, which is also described in the first paragraph of the cover letter. This summary normally includes the following information:

a. Absence of Proprietary Information. At the exit meeting, the inspectors should verify whether or not the licensee considers any materials provided to or reviewed by the inspectors to be proprietary.

NOTE: When an inspection is likely to involve proprietary information (i.e., based on the technical area or other considerations of inspection scope), the topic of how to handle such information should be discussed at the entrance meeting).

If the licensee does not identify any material as proprietary, the exit meeting summary should include a sentence to that effect (see Inspection Manual Chapter (IMC) 0611 on actions to take if the report includes proprietary material). Will be incorporated into Exhibit 2 Section 4AO6.

b. Subsequent Contacts or Changes in NRC Position. The inspector should briefly discuss any contact with the licensee management after the exit meeting to discuss new information relevant to an inspection finding. In addition, if the NRC's position on an inspection finding changed after the exit meeting, that change should be discussed with the licensee before the report is issued.

The following information normally is not included in the exit meeting summary.

c. Characterization of Licensee Response. In general, the report should not characterize a licensee's exit meeting response. If the licensee disagrees with the inspector's finding, this position may be characterized by the licensee in their formal response to the inspection report, if applicable. Specific items discussed elsewhere in the report should not be described in this section in detail.

d. Oral Statements and Regulatory Commitments. If, at the exit meeting or at any other time during the inspection, the licensee makes an oral statement that it will take a specific action, the report should not attempt to characterize that statement nor should this be interpreted as a commitment. Should the licensee wish to make a commitment, the commitment should be documented by licensee correspondence, after which the inspector may reference the correspondence in the inspection report. Oral statements made or endorsed by a member of licensee management authorized to make commitments are not regulatory commitments unless they are documented by the licensee as such. For further guidance on licensee commitments, see ADAMS Accession Nos. ML003680088 (NEI 99-04), ML003680078 (NEI Cover Letter), and ML003679799 (SECY 00-045 endorsing NEI 99-04 guidance).

Because regulatory commitments are a sensitive area, the inspector should ensure that any reporting of such a licensee-documented statement is paraphrased accurately, and contains appropriate reference to the licensee's document.

05.08 Report Attachments. The attachments discussed below are included at the end of the inspection report if applicable to the inspection. The attachments may be combined into a single attachment titled "Supplementary Information."

a. Key Points of Contact. The inspector lists, by name and title, those individuals who furnished relevant information or were key points of contact during the inspection (except in cases where there is a need to protect the identity of an individual). The list should not be exhaustive: a list of 5-10 individuals is sufficient. The alphabetized list includes the most senior licensee manager present at the exit meeting and NRC technical personnel who were involved in the inspection if they were not listed as inspectors on the cover page.

b. List of Items Opened, Closed, and Discussed. The report should provide a quick-reference list of items opened and closed, including the item type, the tracking number for the item, and a brief phrase matching the title used in PIM headers describing the item. Open items that were discussed (but not closed) should also be included in this list, along with a reference to the sections in the report in which the items were discussed. Will incorporate into the sample list included with Exhibit 2.

c. List of Documents Reviewed. A listing of the documents and records reviewed during an inspection is to be publicly available. Therefore, if a listing is not otherwise made public, the report should include a listing of all the documents and records reviewed during the inspection that are not identified in the body of the report. (Reference IMC 0620 Inspection Documents and Records). "Reviewed" in this context means to examine critically or deliberately. The list does not include records that were only superficially reviewed.

d. List of Acronyms. Reports whose details section exceeds 20 pages in length must include a list of acronyms as an attachment. For reports in which a relatively small number of acronyms have been used, the list is optional. In all cases, however, acronyms should be clearly defined when first used in inspection report text.

05.09 Release and Disclosure of Inspection Reports

a. General Public Disclosure and Exemptions. Except for report enclosures containing exempt information, all final inspection reports will be routinely disclosed to the public. IMC 0611, "Review and Distribution of Inspection Reports," describes the various types of exempt information. IMC 0620, "Inspection Documents and Records," gives guidance on acquiring and controlling NRC records, including inspection-related documents.

b. Release of Investigation-Related Information. When an inspector accompanies an investigator on an investigation, the inspector must not release either the investigation report or his or her individual input to the investigation report. This information is exempt from disclosure as required by 10 CFR 9.5, and must not be circulated outside the NRC without specific approval of the Chairman (refer to OI Policy Statement 23).

0610-06 Guidance Other

06.01 Thresholds of Significance. This section gives guidance on how to determine if violations and issues rise to a level of significance that warrant documentation, and on when and how to document findings related to cross-cutting issues.

Two paths lead to documenting findings or violations. One path processes an issue through the SDP and ends in a finding with a color designating an associated safety significance. For example:

A maintenance rule issue about unreliability and unavailability of a high pressure safety injection (HPSI) pump, which affects the functionality of a mitigating system would have its risk characterized by being evaluated by the SDP, after which the issue becomes a finding and is assigned a color to characterize the safety significance.

The second path addresses issues that either (1) are of more than minor significance but are not related to a cornerstone, or (2) are minor issues with extenuating circumstances. If this path is more suitable to the issue, the issue becomes a finding without an assigned color, and the safety significance is related to with the severity levels in the NRC Enforcement Manual. For example:

> A maintenance rule issue regarding failure to perform the annual/refueling evaluation pursuant to 50.65 (a)(3), and failed to have several risk significance systems within scope of the rule would not be suitable for the SDP. However, this would likely be a no color Severity Level IV violation and processed in accordance with the NRC Enforcement Manual.

Each path asks a final question: "Is the finding a violation?" If the finding was assessed using the SDP and is a violation, then it has a color defining the safety significance associated with it. If the finding resulted from being an extenuating circumstance and is a violation, it has no color and its significance is designated by the severity level using the Enforcement Policy. In either case the issue is documented.

The documenting screening process (Appendix B) uses three sets of screening questions and a flow diagram. The questions are intended to (1) assure all significant issues are documented, and (2) increase the consistency of issues NRC inspectors document. Inspectors should use Appendix B Figure 1 and group 1,2 and 3 questions in determining whether an issue should be documented in an inspection report. The decision points in this process are discussed in detail below:

a. Issues. The inspector identifies a concern believed to constitute an issue. The inspector must then determine whether the issue warrants further analysis by the SDP or whether the issue is minor.

b. Minor Issue/Violation (Group 1 Questions). The inspector uses Appendix B group 1 questions to determine if an issue can be considered minor. If, after considering group 1 questions, the inspector cannot decide if the issue is minor or not, the inspector should refer to the NRC Office of Enforcement (OE) "Guidance for Classifying Violations as Minor Violations," for additional guidance. This document is on the OE's WEB-page under Guidance Documents, Appendix A, and Index, "Guidance for Classifying Violations as Minor Violations." If the finding does not have more than minor significance, it should not be documented.If the answer to any group 1 question is "Yes", the issue is considered to be more than minor. The inspector should then determine if the issue affects a cornerstone by asking Appendix B group 2 questions. If the answer to all the group 1 questions is "No", the issue can be considered minor. However, the inspector should also review the group 3 questions to determine whether the issue has extenuating circumstances which may warrant documenting the issue.

   Documenting a minor violation may be necessary in several circumstances such as (1) closing a licensee event report, or (2) information relates directly to an issue of agency-wide concern (e.g., in documenting the results of an NRC temporary instruction). If the inspector determines that it is necessary to document a minor issue which is also a violation, then the inspector documents it as a minor violation and references Section IV of the NRC Enforcement Policy, NUREG-1600, such as: "Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC'sEnforcement Policy." Minor violations are not included in the Summary of Findings or the cover letter and are not given a tracking number.

c. Issues Affecting Cornerstones (Group 2 Questions). The SDP evaluates safety significance and assign colors to those issues which affect a cornerstone. Appendix B group 2 questions should be used to determine whether an issue affects a cornerstone.

   If the answer to any group 2 question is "yes", the issue should be analyzed using the SDP process, documented in the inspection report and assigned a color. If the answers to all group 2 questions are "no" then the inspector should determine whether there are extenuating circumstances by reviewing group 3 questions which may then merit documenting the issue.

d. Extenuating Circumstances (Group 3 Questions). If an issue is either minor, or more than minor and does not affect a cornerstone, there should be extenuating circumstances associated with the issue in order for it to be documented. Appendix B group 3 questions should be used to determine whether an issue has extenuating circumstances. If all the answers to the group 3 questions are "No", the issue does not have extenuating circumstances and should not be documented. If the answer to any group 3 question is "yes", the issue should be documented as a finding or as a violation. Since the issue/violation did not go through the SDP, a color associated with its safety significance cannot be assigned. All violations greater than minor not assessed using the SDP will be assessed through the enforcement policy for assignment of a severity level.

e. SDP Analysis. All NRC identified findings or violations that have greater than minor significance and are related to a cornerstone, should be documented with a safety significance color assigned to them after evaluation by the SDP.

f. Violations. The SDP assigns findings a safety significance color whether it is a violation or not. All documented violations, either with or without a color, are dispositioned according to the requirements in the Enforcement Policy. Note: Violations that were identified by the licensee, have been previously entered into their corrective action system, and are of very low significance or meet the criteria of Section IV of the Enforcement Policy should not be documented in the cover letter or the summary but should be listed in section 4OA7.

06.02 Issues Related to Cross-Cutting Areas

a. Single Findings. When a finding is evaluated as being more than minor and the cause of the finding is related to one of the three cross-cutting areas of Problem Identification and Resolution, Human Performance, or Establishment of a Safety Conscious Work Environment, the cross-cutting nature of the finding should be described in the inspection report. Pertinent cross-cutting aspects of the finding should be documented along with the inspector's description of the SDP evaluated finding as a contributing or direct cause of the finding, as appropriate. The significance of the finding is determined by the SDP. Inspectors should ensure that the cross-cutting aspects are highlighted in the inspection report description and the summary of findings. Such issues that are related to the cross-cutting area of Problem Identification and Resolution should also be captured in Section 4OA2 of the report to aid in the integration of PI & R issues during the annual IP 71152 inspection. Issues that are associated with a finding that filters out as minor after being subjected to the analysis of group 1 and group 3 questions should not be documented.

b. Multiple Findings. Multiple findings that have a common cause associated with one of the three cross-cutting areas should be first identified as individual findings based on the SDP evaluation. Then, the inspector may consider the accumulation of these findings to constitute a significant cross-cutting issue. The following guidance applies to documenting significant cross-cutting issues that are associated with multiple findings:

(1) Each of the individual findings with which the cross-cutting issue is related must have greater than minor significance.

(2) The cross-cutting issue must have been documented as part of a number of individual findings in either the current or previously issued in the past 12 months reports (sections and previous report numbers must be referenced) and should be associated with more than one cornerstone.

(3) Multiple findings that indicate performance trends or patterns of a significant cross-cutting nature should be documented under either Section 4OA4 or 4OA2. The causally linked relationships of each of the findings and the potential safety impact of the combined affect within the applicable cross-cutting area should be addressed. The results of this effect will be considered a "finding." For

example:

"A performance trend appears to be developing in several cornerstone areas with maintenance errors being the common element. Where as; (1) nine months prior to this inspection maintenance personnel improperly installed a bearing during the refurbishment of the containment spray pump causing the pump to be inoperable (NCV 50-000/00-09-06), (2) six months ago maintenance personnel caused a plant trip during the calibration of the pressurizer pressure transmitter, (finding (50-000/00-12-02), (3) 2 months ago maintenance personnel misaligned the HPSI pump causing its inoperability (NCV 50-000/00-13-04), and (4) during this reporting period maintenance personnel caused a spurious actuation of the safety injection while trouble shooting an emergency diesel generator problem (finding 50-000/00-14-01). The causal relationships of these errors was that some of the maintenance was performed by unqualified technicians. The inspector noted that maintenance staffing on the back shifts was reduced at the completion of the last refueling outage ten months ago which may have contributed to this apparent trend". These individual findings each have had a direct impact on safety, increasing the frequency of initiating events and affecting the reliability, operability and functionality of a train of mitigating equipment. This performance trend is considered a substantive cross-cutting issue not captured in individual issues indicating a performance trend, and is a finding 50-000/00-15-04 characterized as "no color".

Emphasis should be placed on any significant trends or patterns which may be emerging in the different cross-cutting areas. These trends or patterns should be highlighted in the summary of findings. Only a succinct reiteration of the common theme is necessary. The finding should then be carried forward in the PIM and coded as "Miscellaneous" vice a specific cornerstone and the significance should be "not applicable."

c. Programmatic Issues within Cross-cutting Areas. Many of the licensee's programs related to maintaining the condition and operability of System Structures and Components (SSCs) are in effect, elements of the licensee's problem identification and resolution program. Therefore, when assessing the impact of Maintenance Rule or other programmatic deficiencies, the finding must include consideration of any equipment failures that were impacted by the deficient programmatic area. The significance of the finding, including the programmatic deficiency is determined by the impact of the equipment failures within the applicable cornerstone. If the programmatic deficiency has no impact on a cornerstone it cannot be assessed using the SDP and therefore, if greater that minor, would be subject to the group three questions and could result in a "No Color" finding. However, these findings should be carefully scrutinized for being potentially minor.

06.03 Documenting Noncompliances. The primary guidance for all matters related to enforcement, including documentation, is in the NRC Enforcement Policy (NUREG-1600) and in Section 3.12 of the NRC Enforcement Manual (NUREG/BR-0195).

The guidance in the Enforcement Policy and Manual applies to issues found or reviewed during inspections that are also violations of regulatory requirements. The SDP will be used, where applicable, for making the determination of significance. Issues that are not evaluated under the significance determination process and those that should be considered for civil penalties will be processed in accordance with the Enforcement Policy. Such issues are typically situations with actual safety consequences (such as an overexposure to the public or plant personnel or a substantial release of radioactive material) or are violations related to willfulness or to impeding the regulatory process (such as violations of reporting requirements). See Section 3.5 of the Enforcement Manual.

a. Specific Enforcement Related Guidance. Findings that are minor violations should not be documented but should be discussed with the licensee during the exit meeting following the inspection if not previously discussed. For additional guidance on minor violations refer to Section IV of the NRC

Enforcement Policy, NUREG-1600, "Guidance for Classifying Violations as Minor Violations."

1. Violations that are identified by the NRC and have subsequently been incorporated into the licensee's corrective action program which are determined to be of very low safety significance or are categorized as Severity Level IV will normally be treated as Non-Cited Violations (NCVs) in accordance with the Enforcement Policy. Notices of violations (NOVs) are issued if the violation meets any one of the applicable criteria in Section VI.A of the Enforcement Policy.

   The discussion in the body of the report should include sufficient information to support the conclusion that the issue is more than minor and is a violation of regulatory requirements (regardless of whether the issue will be dispositioned as an NCV or an NOV). At a minimum the report should state:

   - what requirement was violated
   - how the violation occurred
   - when the violation occurred, and how long it existed
   - when the violation was identified
   - any actual or potential safety consequence
   - the root cause (if identified)
   - all information required to complete the SDP
   - what corrective actions have been taken or planned. (For licensees with adequate corrective action programs, it is acceptable to only verify that the licensee has entered the issue in its corrective action program for issues that are of very low significance (Green)).

   A conclusion that the violation will or will not be cited should be documented in the details section of the report. See the language in the Enforcement Manual.

2. For issues that are determined to have more than very low safety significance (i.e., White, Yellow, or Red), in addition to the guidance contained in 05.06.b, should include the following if available at the time of documentation:

   - The assumptions used by the inspector or regional Senior Reactor Analyst (SRA) in determining the finding's significance.
   - The significance attributed to the finding by the licensee and, if different than the NRC's determination, a description of the assumptions used by the licensee, and what the licensee considered applicable to its determination that was different from the NRC's.
   - Pertinent accident sequences and mitigating capabilities.
   - Actions the licensee has taken or plans to take to correct the condition and underlying root cause(s), including the appropriate condition report used to enter the issue into the licensee's corrective action program.
   - The licensee's position on the NRC's determination that a requirement has been violated, if so determined.

   The final significance determination will be documented, the issue entered into the Plant Issues Matrix (PIM), and the associated enforcement action will be taken based on the significance. If the finding is Green, a Violation should be documented in an inspection report, and if the finding is White, Yellow, or Red, a notice of violation will be issued in accordance with the Enforcement Policy.

3. Some issues may have a preliminary significance of greater than Green, for which the safety characterization may not have been finalized at the date of the report issuance. Issues initially categorized as having a potential safety significance of greater than very low significance (Green) but whose significance has not yet been determined should be documented in the report, and the summary of findings with a significance characterization of To Be Determined (TBD). The issue

may be documented as an "apparent violation" if a violation of requirements is associated with the issue, and with a significance of "TBD" in IR. Emphasis should be placed on the safety characterization as being potential and not yet finalized. After a final safety characterization is determined by the SDP oversight and enforcement panel and a letter is sent to the licensee regarding this characterization, the PIM should be updated to reflect the final safety characterization and the next subsequent resident inspector inspection report should include a brief description of the issue and the change in safety classification in the summary of findings.

• Inspectors must be careful to avoid making direct statements regarding safety significance in the inspection report details outside the SDP analysis or for issues not subject to the SDP. Violation severity levels, as described in the NRC Enforcement Policy, are based on the degree of safety significance involved. In addition, the NRC Enforcement Policy uses the term "safety significance" in a specific sense. Inspectors should refer to the NRC Enforcement Manual for the most recent guidance.

• Inspection reports should not solely refer to a noncompliance as being (just) "of very low safety significance."

UNACCEPTABLE: "The issue was determined to be Green by the significance determination process,"

• The inspector should state why that determination was reached.

ACCEPTABLE: "The issue was determined to be of very low significance (Green) by the significance determination process because even though the equipment was degraded it was capable of performing its safety function, and trained operators were also available and ready to take appropriate manual actions if needed."

4. Violations of requirements that cannot be evaluated with the SDP should be documented in the report section relating to the inspectable area in which the violation was discovered, or in Section 4, "Other Activities," if unrelated to a specific inspectable area. The severity level of such violations will be determined using the guidance in the Enforcement Policy and Enforcement Manual.

b. Licensee Identified Violations. Frequently inspectors review issues that have been identified by the licensee and entered into their corrective action program. This is expected in a risk-informed inspection program that attempts to focus inspectors on those issues of potential risk significance. If after examining such licensee identified issues the inspector recognizes that the licensee has correctly evaluated the issue and has developed appropriate corrective actions, and the issue is recognized as being of very low significance, such issues should be referenced in the inspection report for tracking purposes only and not included in the summary of findings or the transmittal cover letter. However these NCVs will be separately captured in the PIM. This is appropriate because it encourages licensee's to self-identify and correct problems. Conversely, inspectors may identify additional deficiencies or concerns associated with the licensee identified issue. In these cases it is appropriate for additional detail regarding the deficiencies to be documented in the inspection report as the inspector has provided value-added in further defining the issue.

Except as noted below violations that are licensee identified which have been incorporated into the licensee's corrective action program and are recognized as very low safety significance or would be categorized as a potential Severity Level IV violation, will normally be only briefly documented in section 40A7 of the inspection report. The documentation must include the NRC tracking number, the requirement violated, a one sentence description of how the requirement was violated and a reference to the licensee corrective action program tracking number or condition report number. For example:

4OA7	Licensee Identified Violations. The following findings of very low significance were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as Non-Cited Violations (NCV).
NCV Tracking Number	Requirement Licensee Failed to Meet
(1) NCV 999/00007-2	Technical Specifications 3.1.A requires three NI channels to be operable during core alterations. Only two channels were operable during core alterations on January 4, 2000, as described in the licensee corrective action program Reference CAP XXX-000/123.

Among these types of issues are those that are discovered during the review of LERs or while inspecting licensee corrective action programs or similar types of inspections.

If the inspector identifies a deficiency with the licensee's evaluation, corrective actions or other problems associated with the licensee identified finding, the inspector should document the finding under section 4OA2 of the report irrespective of who identified it. Documentation should clearly emphasize that the licensee identified the finding but failed to recognize the deficiency or the nexus to the problem identified by the inspector.

c. Noncompliances Involving Willfulness. Inspection reports should neither speculate nor reach conclusions about the intent behind a violation, such as whether it was deliberate, willful, or due to careless disregard. The report should include relevant details on the circumstances of the violation without making a conclusion about the possible intent of the violator:

APPROPRIATE: "The radiographer failed to activate his alarming dosimeter, although he had informed the inspectors earlier that he had been properly trained on the use of the device."

INAPPROPRIATE: "The radiographer deliberately failed to activate his alarming dosimeter."

Conclusions about the willfulness of a violation are agency decisions, and are normally not made until after the Office of Investigation (OI) has completed an investigation. A premature or inaccurate discussion of the willfulness of an apparent violation in the inspection report could result in later conflicts based on additional input and review. Inspection reports that include potentially willful violations are to be coordinated with OI and the Office of Enforcement (OE).

06.04 Treatment of Open Items. Issues that require additional inspection before coming to closure on the issue are identified by a unique tracking number and entered into the Item Reporting (IR) module of the Reactor Programs System (RPS) by the originating inspector or office. Open items include unresolved items, violations, deviations, licensee event reports (LERs), and SDP-related issues whose significance has yet to be determined. NCV follow-up is limited to sampling when assessing the licensee's corrective action program.

a. Initiating Open Items. The action of initiating an open item is a commitment of future resources, and should therefore only be used when some specific licensee action is pending, or when needed information is not available at the time of the inspection. When the inspector believes that the additional information may reveal the issue to be a matter of noncompliance, an unresolved item should be initiated. For an unresolved item, the report should identify the actions or additional inspection effort needed to resolve the issue.

Issues of noncompliance (except for minor violations) should always be assigned an inspection report item number for tracking purposes. When an inspection involves multiple violations (or multiple examples of a single violation), the inspector should be careful to ensure a one-to-one correlation between the number of IR entries and the number of "contrary to" statements in the accompanying notice of violation. The NRC Enforcement Manual provides additional guidance on tracking and following up issues of noncompliance.

Upon receipt, LERs should be entered into the IR module system for tracking, screening and follow-up.

b. Follow-Up and Closure of Open Items. The level of detail devoted to closing open items depends on the nature and significance of the additional information identified. The closure of an open item should, at a minimum, summarize the topic, summarize the inspector's follow-up actions, evaluate the adequacy of any licensee actions, determine if a violation occurred, and include enough detail to justify closing the issue.

   The close out description of a violation should be brief if the licensee's response to the notice of violation already has given an accurate description of the root cause, corrective actions taken, and other aspects of the condition causing the violation, and the inspector identifies no other instances of the violation. Normally NCVs will be opened and closed in the initiating inspection report.

c. Treatment of Events and Licensee Event Reports. Followup of events and LERs are addressed in several areas including IMC 2515 "Light-Water Reactor Inspection Program," IP 71153 "Event Follow-up", and IP 71111.14 "Personnel Performance During Non-Routine Plant Evolutions". Each requires that all LERs be at least screened by an inspector and closed in an inspection report. LERs are initially screened and can be closed after an in-office review based upon the inspector's engineering judgment. Those LERs determined to involve complex events, are immediately recognized as greater than very low significance, events which caused a performance indicator to exceed a threshold, or as directed by one of the above procedures should be considered for follow-up inspection at the facility. Events and LER discussions, including revisions to LERs, should be documented in the inspection report under Section 4OA3, "Event Follow-up". If inspection in another cornerstone area provides a description of an event, or an event for which an LER is issued (i.e., personnel performance during non-routine evolutions), that section of the report should be referenced under Section 4OA3 with a very brief description. (Example will be incorporated in Exhibit 2 Section 4OA3). In general LER reviews should have a brief event description, reference the docketed LER, and require little discussion other than the significance evaluation and reference to the licensee's corrective action program (CAP) system tracking number for the issue.

   For LERs involving minor findings, potential violations meeting the criteria for being minor, or issues that the licensee identified, entered into their corrective action program and are of very low significance, the LER closure documentation should note that the issue is captured in the licensee's corrective action program, reference the LER, and state that the LER was reviewed and no findings of significance were identified. LERs that were already addressed by separate NRC letter should also be closed with a brief statement in an inspection report.

   When the LER involves more than a minor issue, the inspection report should describe the safety significance of the event, the corrective actions (referencing the (CAP) tracking number), the licensee's determination of the apparent cause, a summary of the inspector's follow-up actions, and any required enforcement actions. If a special inspection was conducted which would provide additional information regarding this event, the inspection report should be referenced.

   LERs frequently involve violations of TS or other requirements. If the LER states a violation occurred the violation must be clearly identified in the report as a cited violation, a noncited violation, or a minor violation, as appropriate. (Otherwise, a statement should be included that "this event did not constitute a violation of NRC requirements.") This should be the last statement of the Section.

   If an LER describes an issue which may be a potential violation and readily appears to be of no more than very low safety significance, the inspector should ascertain if a noncompliance occurred based on the inspectors knowledge of NRC regulations and the content of the LER, without necessarily gathering additional details. Depending on the details of the issue, the inspector should document the issue in the inspection report as described above referencing the licensee's corrective action program tracking number. If the issue is determined to be greater than very low significance, a more detailed onsite follow-up is required if not already performed.

d. Avoiding "Implied" Inspection Follow-up Items. Other than what is implied in discussing open items, the inspection report should not commit to future NRC attention in a particular area. This will be part of inspection planning and the assessment process described in IMC 0305.

e. Documenting Performance Indicators (PIs). Performance indicator inspection should be documented under section 40A1 in the inspection report. The scope section should include the period of time for which the data was reviewed. Data reported prior to January 2000 is considered historical data and should not be reviewed. The criteria used to verify the PI should be included, (Example to be incorporated into Exhibit 2 Section 4OA1). List the PIs verified and the associated cornerstones. When there are three or more PIs being verified, the scope and findings can be listed separately for each if there are findings.

The findings Section should include those occurrences that would cause a PI to cross a threshold. Minor issues should not be documented unless the issue results in reporting inadequacies or interpretations related to the current version of the NEI 99-02 guidance.

Interpretation issues should be briefly described and captured as an URI - "The resolution of this item is pending a response from Headquarters. It is identified as URI 50-XXX/YYY."

f. Treatment of Third Party Reviews. Reviews of Institute of Nuclear Power Operations (INPO) evaluations or accreditation reports or similar third party reviews that identify confidential safety issues should be documented under 4OA5. This should be a short statement stating that the review of a specified evaluation or accreditation was completed. Documenting an INPO evaluation or accreditation report review, should not include a recounting or listing of INPO findings or reference a final INPO rating. Specifics of any significant differences between NRC and INPO perceptions should be discussed with regional management.

END

EXHIBITS:

Exhibit 1: Standard Reactor Inspection Report Outline

Exhibit 2: Sample Reactor Inspection Report

APPENDICES:

Appendix A: List of Acronyms Used in IMC 0610*

Appendix B: Thresholds for Documentation

Appendix C: Documentation Guidance for Supplemental Inspections

Appendix D: Guidance For Documenting Inspection Procedure 71152 "Identification and Resolution of Problems"

STANDARD REACTOR INSPECTION REPORT OUTLINE Exhibit 1

Cover Letter

Cover Page

Summary of Findings

Table of Contents (optional)

Report Details:

1 REACTOR SAFETY

Initiating Events/Mitigating Systems/Barrier Integrity [REACTOR - R]

Note: The baseline inspection procedure number is provided here as a convenience. It may be added to the headings in inspection reports at the option of the region.

[Number	Topic	Baseline Procedure]
R01	Adverse Weather Protection	71111.01
R02	Evaluation of Changes, Tests, or Experiments	71111.02
R03	[R03 Reserved]	
R04	Equipment Alignment	71111.04
R05	Fire Protection	71111.05
R06	Flood Protection Measures	71111.06
R07	Heat Sink Performance	71111.07
R08	Inservice Inspection Activities	71111.08
R09	[R09 Reserved]	
R10	[R10 Reserved]	
R11	Licensed Operator Requalification	71111.11
R12	Maintenance Rule Implementation	71111.12
R13	Maintenance Risk Assessments and Emergent Work Evaluation	71111.13
R14	Personnel Performance During Non-routine Plant Evolutions	71111.14
R15	Operability Evaluations	71111.15
R16	Operator Work-Arounds	71111.16
R17	Permanent Plant Modifications	71111.17
R18	[R18 Reserved]	
R19	Post-Maintenance Testing	71111.19
R20	Refueling and Outage Activities	71111.20
R21	Safety System Design and Performance Capability	71111.21
R22	Surveillance Testing	71111.22
R23	Temporary Plant Modifications	71111.23
Emergency Preparedness [EP]		
EP1	Exercise Evaluation	71114.01
EP2	Alert Notification System Testing	71114.02
EP3	Emergency Response Organization Augmentation Testing	71114.03
EP4	Emergency Action Level and Emergency Plan Changes	71114.04
EP5	Correction of Emergency Preparedness Weaknesses and Deficiencies	71114.05
EP6	Drill Evaluation	71114.06

2. RADIATION SAFETY

Occupational Radiation Safety [OS]

OS1	Access Control to Radiologically Significant Areas	71121.01
OS2	ALARA Planning and Controls	71121.02
OS3	Radiation Monitoring Instrumentation	71121.03
Public Radiation Safety [PS]		
PS1	Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems	71122.01
PS2	Radioactive Material Processing and Transportation	71122.02
PS3	Radiological Environmental Monitoring Program	71122.03
3. SAFEGUARDS		
Physical Protection [PP]		
PP1	Access Authorization	71130.01
PP2	Access Control	71130.02
PP3	Response to Contingency Events	71130.03
PP4	Security Plan Changes	71130.04
4. OTHER ACTIVITIES [OA]		
OA1	Performance Indicator Verification	71151 (Note 1)
OA2	Identification and Resolution of Problems	71152 (Note 2)
OA3	Event Follow-up	71153 (Note3)
OA4	Cross-cutting Issues	
OA5	Other	(Note 5)
OA6	Meetings, including Exit	
OA7	Licensee Identified Violations	

NOTES:

1. Any findings related to the performance indicator (PI) verification baseline inspection shall be included under Other, 4OA1.
2. Section 4OA2 is to be used to document the annual identification and resolution of problems, IP 71152, significant trends relating to the corrective action process that are exemplified by other documented inspection findings, and to reference findings discussed in cornerstone areas related to PI&R issues.
3. Section 40A3 is to be used to discuss both following up on recent events using Inspection Procedure 71153 and reported events (LERs). Discussions in other cornerstone areas which provide a description of an event for which an LER is issued should also be referenced under 4OA3.
4. Section 4OA4 is to be used only to document significant trends in the cross-cutting areas.
5. Reviews conducted of Institute of Nuclear Power Operations (INPO) and third party evaluations are included in Section 4OA5.

END

SAMPLE REACTOR INSPECTION REPORT Exhibit 2

NOTE: The inspection report that follows is based on a fictional reactor licensee and a fictional inspection. The report contains realistic issues; however, any resemblance to an existing facility or actual events is coincidental.

This exhibit may be used as a sample report for format and style. It illustrates how to use the standardized inspection report outline, and adheres to the expected internal organization for each report Section (as discussed in IMC 0610).

Pages are numbered continuously through this exhibit. Inspection reports should use separate page numbering for the cover letter, summary of findings, and report details. Note that these will be provided at a later date when experience is gained with this version of IMC 0610*.

SAMPLE COVER LETTER NO.1 (No Findings)

August 14, 1999

Ms. Joan A. Doe, Vice President, Nuclear

Greckenshire Power & Light

721Y Brick Road

Stone Towers, WF 44632

SUBJECT: DIROJAC GENERATING STATION- NRC INSPECTION REPORT 50-998/99-07, 50-999/99-07

Dear Ms. Doe:

On July 24, 1999, the NRC completed an inspection at your Dirojac Units 1 and 2. The enclosed report documents the inspection findings which were discussed on July 24, 1999, with Mr. D. Prue and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

No findings of significance were identified.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

Projects Branch 8
Division of Reactor Projects

Docket Nos.: 50-998, 50-999

License Nos: XXX-77, XXX-79

Enclosure: Inspection Report 50-998/99-07, 50-999/99-07

Attachments: (1) Supplemental Information
(2) List of Documents Reviewed
(3) List of Acronyms Used

cc w/encl: L. Collinsworth, Compliance Manager
R. Littleroy, General Manager, Technical Services
J. Bradwood, Plant General Manager
F. Buckfry, General Counsel
D. Soapsam, Operations Manager

SAMPLE COVER LETTER NO. 2 (White/Yellow/Red ISSUE)

EA-YY-XXX

Licensee Address

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A (WHITE, YELLOW, RED) FINDING (if applicable, add: "AND NOTICE OF VIOLATION")(NRC Inspection Report No(s). XX-XXX/YY-NN)

(include name of facility)

Dear (Licensee Official):

The purpose of this letter is to provide you with the final results of our significance determination of the preliminary (White, Yellow, Red) finding identified in the subject inspection report. Inspection finding(s) were assessed using the significance determination process and were preliminarily characterized as (White, Yellow, Red), (i.e., an issue with low to moderate increased importance to safety, which may require additional NRC inspections, an issue with substantial importance to safety that will result in additional NRC inspection and potentially other NRC action; (red) an issue of high importance to safety that will result in increased NRC inspection and other NRC action). This (White, Yellow/Red) finding involved (describe the findings).

[For declination of a regulatory conference, include the following paragraph:]

In a telephone conversation with Mr. ___ of NRC, Region X, on Date, (responsible Licensee) of your staff indicated that (Licensee) did not contest the characterization of the risk significance of this finding and that you declined your opportunity to discuss this issue in a Regulatory Conference.

[For regulatory conferences, include the following paragraph:]

At your request, a Regulatory Conference was held on (Date), to further discuss your views on this issue. (A copy of the handout you provided at this meeting is attached.) During the meeting your staff described your assessment of the significance of the findings, detailed corrective actions, including the root cause evaluations for the event classification issues. Specifically, (provide additional details of the licensee assessment if needed).

After considering the information developed during the inspection (if applicable, add: "the additional information you provided in your letter dated (date), and the information you provided at the conference"), the NRC has concluded that the inspection finding is appropriately characterized as (White, Yellow, Red), (i.e., an issue with low to moderate increased importance to safety, which may require additional NRC inspections, an issue with substantial importance to safety that will result on additional NRC inspection and potentially other NRC action; an issue of high importance to safety that will result in increased NRC inspection and other NRC action).

You have 10 business days from the date of this letter to appeal the staff's determination of significance for the identified [white/yellow/red] finding[s]. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Supplement 3.

The NRC has also determined that (describe the violation) is a violation of (list the requirement), as cited in the attached Notice of Violation (Notice). The circumstances surrounding the violation are described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice of Violation is considered escalated enforcement action because it is associated with a (White, Yellow, Red) finding.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

Because plant performance for this issue has been determined to be in the ____ regulatory response band, we will use the NRC Action Matrix, to determine the most appropriate NRC response for this event. We will notify you, by separate correspondence, of that determination.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,
Regional Administrator (or designee)

Docket Nos: 50-99X, 50-9X9

License Nos: XXX-77, XXX-77,

Enclosure: Report No. 05000xxx/1999-007, 05000xxx/1999-007

Attachments: (1) Supplemental Information
(2) List of Documents Reviewed
(3) List of Acronyms Used

SAMPLE COVER LETTER NO.3 (Green Issue and NCVs)

August 14, 1999

Ms. Joan A. Doe, Vice President, Nuclear

Greckenshire Power & Light

721Y Brick Road

Stone Towers, WF 44632

SUBJECT: DIROJAC GENERATING STATION- NRC INSPECTION REPORT 50-998/99-07, 50-999/99-07

Dear Ms. Doe:

On July 24, 1999, the NRC completed an inspection at your Dirojac Units 1 and 2. The enclosed report documents the inspection findings which were discussed on July 24, 1999, with Mr. D. Prue and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified three issues of very low safety significance (Green). Two of these issues were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as Non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these noncited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region ___; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Dirojac facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,
Projects Branch 8
Division of Reactor Projects

Docket Nos.: 50-998, 50-999

License Nos: XXX-77, XXX-79

Enclosure(s):Inspection Report 50-998/99-07, 50-999/99-07

Attachments: (1) Supplemental Information
(2) List of Documents Reviewed
(3) List of Acronyms Used

cc w/encl: L. Collinsworth, Compliance Manager

R. Littleroy, General Manager, Technical Services

J. Bradwood, Plant General Manager

F. Buckfry, General Counsel

D. Soapsam, Operations Manager

EXAMPLE INSPECTION REPORT

U.S. NUCLEAR REGULATORY COMMISSION

REGION X

Docket Nos: 50-998, 50-999

License Nos: XXX-77, XXX-79

Report No: 50-998/99-07, 50-999/99-07

Licensee: Greckenshire Power & Light (GP&L)

Facility: Dirojac Generating Station, Units and 2

Location: 11555 Granite Blvd.

Stone Towers, WF 44632

Dates: June 11-July 24, 1999

Inspectors: A. Rand, Senior Resident Inspector

M. Heidegger, Resident Inspector

J. Locke, Senior Radiation Specialist

P. Sappho, Reactor Projects Inspector

Approved by: E. Tudor, Chief, Projects Branch 2

Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000998-99-07, IR 05000999-99-07, on 06/01-07/24/1999, Greckenshire Power & Light, Dirojac Generating Station, Units 1 & 2. Emergent work, equip-alignment, inservice inspection, non-routine plant evolutions, post-maint. testing, refueling & outage.

The inspection was conducted by resident inspectors, a regional radiation specialist, and a regional projects inspector. The inspection identified three Green findings, two of which were noncited violations. The significance of most/ all findings is indicated by their color (Green, White, Yellow, red) using IMC 0609 "Significance Determination Process" (SDP) . Findings for which the SDP does not apply are indicated by "no color" or by the severity level of the applicable violation.

A. Inspector Identified Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a Non-Cited Violation for failure to assure that nondestructive examination contract inspectors were qualified in accordance with ANSI N45.2.6.c.2

  The finding was of very low safety significance because, although the inspector performing the reactor vessel weld inspections was not qualified, a different inspector reperformed the weld inspections and did not identify any significant weld deformities. (Section 1R08).
- Green. During plant startup operators failed to initiate emergency feedwater, resulting in an uncomplicated unit trip. The inspectors identified a Non-Cited violation for inadequate procedures (Technical Specification 6.8.1).

  The safety significance of this finding was very low because all mitigation systems remained operable, barrier integrity was not challenged, and the licensee entered the finding into the corrective action program. (Section 1R14).
- Green. The inspectors identified that the licensee's in-progress corrective actions for failure of a drywell fan did not include resolution of the subsequent increase in drywell temperatures above final safety analysis report limits for drywell snubbers.

  This finding was of very low safety significance because the licensee subsequently determined that the snubbers remained functional, although the increased temperature shortened their life by 1 year (Section 1R03).

Cornerstone: Occupational Radiation Safety

- Green. Radiation protection technicians together with the NRC inspectors identified that the licensee failed to remove all material containing low levels of radioactive contamination from a temporary radiologically protected area (10 CFR 20.xyz.(b)), before the area was released for unrestricted use. Additionally the licensee had recently identified two similar problems in radiological problem reports.

  The finding was of very low safety significance because the contamination did not spread beyond the radiological area and the licensee identified and corrected the problem. The inspectors identified this as a Non-Cited violation for failing to follow procedures. (Section 2OS1).

Cross-cutting Issues: Human Performance

- No Color. Similar human performance errors were identified in both initiating event and mitigating system cornerstone areas. Inspectors found that errors in review, coordination, and implementation of maintenance activities during or near Unit 2 refueling outage number 12 led to inoperable safety systems. Operators were unaware that Technical Specification (TS) or administrative limiting-condition-for-operation action statements were entered or exceeded. Required nuclear instruments and emergency diesel generators were not operable during fuel moves (50-998/99-06 Sections 1R04.2 and 1R20.4), automatic depressurization system valves were taken out of service while required (Section 1R20.2), and the high- pressure coolant injection system was inoperable because of incomplete maintenance (50-998/99-05 Section 1R19.1).

  While the risk of the individual events was very low, the number of maintenance-related incidents indicated a performance trend of problems with control, review, and performance of maintenance activities (Section 1R20).

B. Licensee Identified Violations

Violations of very low significance which were identified by the licensee have been reviewed by the inspector. Corrective actions taken or planned by the licensee appear reasonable. These violations are listed in section 4OA7 of this report.

EXAMPLE 2 IS UNDER DEVELOPMENT

Appendix A : List of Acronyms Used in IMC 0610*

NOTE: a separate list of acronyms is given as an enclosure to Exhibit 2, the sample inspection report.

AEOD Office for Analysis and Evaluation of Operational Data

ALARA as low as is reasonably achievable

CFR Code of Federal Regulations

CVCS chemical and volume control system

EA escalated action

EP emergency preparedness

ESF engineered safety feature

EW exercise weakness

gpm gallons per minute

GPO Government Printing Office

IFI inspection follow-up item

IFS Inspection Follow-Up System

IMC inspection manual chapter

IPAP Integrated Performance Assessment Process

IR Item Reporting Module

ISI in-service inspection

LER licensee event report

LOCA Loss-of-Coolant Accident

MD management directive

MREM Milli-roentgen equivalent man

NCV noncited violation

NMSS Office of Nuclear Material Safety and Safeguards

NOV notice of violation

NRC Nuclear Regulatory Commission

NRR Office of Nuclear Reactor Regulation

OE Office of Enforcement

OI Office of Investigations

PIPB Inspection Program Branch

PPR plant performance review

PRA probabilistic risk assessment

RA regional administrator

RHR residual heat removal

RP radiation protection

RP&C radiological protection and chemistry

SDP Significance Determination Process

SI International System of Units

TBD to be determined

TI temporary instruction

TS technical specification

Appendix B : Thresholds for Documentation

Inspectors use Figure 1 and group 1, 2, and 3 questions in determining if an issue should be documented in an inspection report. The decision points in this process are discussed in detail below. For all the below questions, "could" refers to application of credible scenarios.(see definitions).

A. Issues

The inspector identifies an issue. The inspector should first determine whether the issue has sufficient significance to warrant further analysis or documentation. This is done by determining whether the issue is minor. Minor issues should not be documented in inspection reports.

B. Minor Issues/Violations (group 1 questions)

If the answer to any of the below questions is "Yes", the issue can be considered greater than minor and the inspector should review group 2 questions to determine if the issue impacts a cornerstone. If the answers to all of the group one questions is "No", the issue may be considered minor. However, the inspector should also determine whether the issue has extenuating circumstances that warrant documenting the issue in the inspections report by reviewing group 3 questions. Additional guidance and examples can be found in the NRC Enforcement Manual, Guidance Documents, "Guidance for Classifying Violations as Minor Violations."

Group 1 Questions

Group 1 questions are intended to parallel the Enforcement Manual's guidance on what constitutes a minor violation. Numerous examples are provided in this guidance for a variety of issues and provide clarity regarding complex issues such as those associated with Maintenance Rule findings. Inspectors should consult this guidance after reviewing group 1 questions if there is any question whether an issue should be considered minor.

(1) Does the issue have an actual or credible impact on safety?

(2) Could the issue be reasonably viewed as a precursor to a significant event?

(3) If left uncorrected, would the same issue under the same conditions become a more significant safety concern?

(4) Does the issue relate to collecting or reporting performance indicators that would have caused a PI to exceed a threshold?

C. Issues Affecting Cornerstones (Group 2 Questions)

If the answer to any group 2 question is "Yes", the issue should be analyzed by the SDP process, assigned a color, and documented in the inspection report. If the answers to all group 2 questions are "No", then the inspector should determine whether there are extenuating circumstances by reviewing the group 3 questions.

(Note: Group 2 questions are intended to determine if the identified issues which impact a cornerstone. "No" only means that the issue is not suitable for SDP evaluation).

Group 2 Questions

Reactor Safety--Initiating Events, Mitigating Systems, & Barrier Integrity

(1) Could the issue cause or increase the frequency of an initiating event?

(2) Could the issue credibly affect the operability, availability, reliability, or function of a system or train in a mitigating system?

(3) Could the issue affect the integrity of fuel cladding, the reactor coolant system, reactor containment or control room envelope?

(4) Does the performance of issue involve degraded conditions that could concurrently influence any mitigation equipment and an initiating event?

Reactor Safety--Emergency Planning

(1) Does the issue involve a failure to meet or implement a regulatory requirement?

(2) Does the issue involve a drill or exercise critique problem?

Radiation Safety--Occupational (ALARA)

(1) Does the actual job dose exceed the projected dose by >50%, AND does the 3-year rolling average collective dose exceed 135 person-rem/unit for a PWR or 240 person-rem/unit for a BWR, AND is the actual job dose > 5 person-rem?

(2) Does the occurrence involve an individual worker(s) unplanned, unintended dose(s) that resulted from actions or conditions contrary to licensee procedures, radiation work permit, technical specifications or NRC regulations?

(3) Does the occurrence involve an individual worker(s) unplanned, unintended dose(s) or potential of such a dose (resulting from actions or conditions contrary to licensee procedures, radiation work permit, technical specifications or NRC regulations) which could have been significantly greater as a result of a single minor, reasonable alteration of the circumstances?

(4) Does the occurrence involve conditions contrary to licensee procedures, technical specifications or NRC regulations which impact radiation monitors, instrumentation and/or personnel dosimetry, related to measuring worker dose?

Radiation Safety--Public

(1) Does the issue involve an occurrence in the licensee's radiological effluent monitoring program that is contrary to NRC regulations or the licensee's TS, Offsite Dose Calculation Manual (ODCM), or procedures?

(2) Does the issue involve an occurrence in the licensee's radiological environmental monitoring program that is contrary to NRC regulations or the licensee's TS, ODCM, or procedures?

(3) Does the issue involve an occurrence in the licensee's radioactive material control program that is contrary to NRC regulations or the licensee's procedures?

(4) Does the issue involve an occurrence in the licensee's radioactive material transportation program that is contrary to NRC or Department of Transportation (DOT) regulations or licensee procedures?

Physical Protection

(1) Does the issue involve a nonconformance with safeguards requirements?

Fire Protection

(1) Does the issue involve impairment or degradation of a fire protection feature?

D. Extenuating Circumstances (Group 3 Questions)

If an issue is either minor or more than minor and does not affect a cornerstone, there should be extenuating circumstances associated with the issue that would warrant documentation of the issue. The following questions in group 3 should be reviewed to determine whether an issue has extenuating circumstances.

(1) Are there any associated circumstances that add regulatory or safety concerns (i.e., apparent willfulness, licensee refusal to comply, or discrimination)?

(2) Does the issue have potential for impacting the NRC's ability to perform its regulatory function? For example, a failure to provide complete and accurate information or to perform 10 CFR 50.59 analyses, etc. (see Enforcement Policy IV.A.3)

(3) Is documenting this issue necessary to close an open item such as a licensee event report?

(4) Does the associated technical information relate directly to an issue of agency-wide concern (i.e., a generic safety issue)?

(5) Does the issue describe a substantive cross-cutting issue which has been captured in a number of individual findings in the current or previous reports or which indicates adverse performance trends or patterns?

(6) Was the issue determined to be a violation greater than minor during the review of group 1 questions?

If all the answers to the above questions are "No", the issue does not have extenuating circumstances and would not normally be documented. If the answer to any question is "Yes", the issue should be documented as a finding or a violation without a color.

Note: Credible scenarios must reflect the actual condition or analysis and may assume only one additional hypothetical condition or failure. For example, under a given condition an accident analysis assumes one passive or one active failure in combination with the degraded condition identified during the inspection. It is not credible to assume a change in those conditions and hypothesize an additional failure. Discussions with "if," "potentially," and "could have" regarding the same issue should be reviewed carefully to ensure the finding is credible.

Thresholds for Documentation

Figure 1

Appendix C : Guidance for Supplemental Inspections

In general, most of the guidance contained in this Inspection Manual chapter applies equally to the baseline and the supplemental portions of the power reactor inspection program. However, due to the nature of the supplemental inspections, it is expected that the associated supplemental inspection reports will contain a more complete documentation of the NRC's assessment of each inspection requirement, including pertinent qualitative observations of the licensee's efforts to identify and address the root cause of the issue. The following guidance applies specifically to the documentation of inspections using supplemental Inspection Procedures 95001 and 95002:

- a separate inspection report will usually be generated for each supplemental inspection
- the inspection report will contain the following Sections:
  - a summary of findings (to be entered into the PIM), which will provide an overall assessment of the licensee's evaluation of the performance issue, including any specific findings associated with the licensee's evaluation, or findings associated with new issues that emerged during the inspection,
  - a summary of the performance issue for which the inspection is being performed (this can be taken from a previous inspection report for a inspection issue or can be a summary of the PI and the particulars associated with its crossing a threshold),
  - restatement of each inspection requirement (or an abbreviated heading describing each requirement), followed by a synopsis of the licensee's assessment related to the inspection requirement, followed by the inspector's assessment of the licensee's evaluation, including a description of any additional actions taken by the inspector to assess the validity of the licensee's evaluation,
  - a list of persons contacted and all licensee documents reviewed during the inspection, and
  - a list of acronyms used in the inspection report.

The independent review of extent of condition called for in Inspection Procedure 95002 and performed using a procedure or procedures chosen from Appendix B to Inspection Manual Chapter 2515 should be documented along with the other inspection requirements contained in Inspection Procedure 95002. Portions of a sample inspection report performed in accordance with supplemental Inspection Procedure 95001 are provided on the following pages. Some Sections of this sample report contain alternative writeups to illustrate how both positive and negative inspection results would be documented.

Specific documentation requirements and report format for supplemental Inspection Procedure 95003 will be provided by the team leader and will generally be similar to that for supplemental Inspection Procedures 95001 and 95002.

U.S. NUCLEAR REGULATORY COMMISSION

REGION X

Docket Nos: 50-998, 50-000

License Nos: xxx-79, xxx-80

Report No: 50-998/2000-08, 50-000/2000-08

Licensee: Iowananuke

Facility: Profit Centers 1 and 2

Location: 1234 Atomic Blvd

Somewhere, USA

Dates: December 25--December 31, 2000

Inspectors: A. Grounder, Senior Resident Inspector

R. Cause, Reactor Projects Inspector

Approved by: S. Slatkin, Projects Branch 1

Division of Reactor Projects

SUMMARY OF FINDINGS

Profit Centers 1 and 2

NRC Inspection Report 50-998/2000-08, 50-000/2000-08

ADAMS TEMPLATE: (TO BE INSERTED HERE, see IMC 0610 Exhibit 2)

Cornerstone: Mitigating Systems

This supplemental inspection was performed by the NRC to assess the licensee's evaluation associated with the inoperability of the Unit 1 diesel generator A. This performance issue was previously characterized as having low to moderate risk significance ("White") in NRC Inspection Report #XXX XXXXX. During this supplemental inspection performed in accordance with Inspection Procedure 95001, the inspectors determined that the licensee performed a comprehensive evaluation of the inoperable diesel. The inoperable diesel was identified by the licensee during a surveillance test. The licensee's evaluation identified the primary root cause of the performance issue to be poor control of vendor manuals, which resulted in the maintenance workers mis-calibrating the governor speed control unit. The vendor manual control issue was not limited to the diesel generator and the licensee has taken corrective actions to ensure vendor manuals are current for all risk significant equipment. In addition, the licensee intends to review the scope of quality assurance audits to determine whether additional resources need to be provided to the quality assurance department to identify similar programmatic deficiencies.

Due to the licensee's acceptable performance in addressing the inoperable Unit 1 diesel generator, the white finding associated with this issue will only be considered in assessing plant performance for a total of four quarters in accordance with the guidance in IMC 0305, "Operating Reactor Assessment Program." Implementation of the licensee's corrective actions will be reviewed during a future inspection.

or

This supplemental inspection was performed by the NRC to assess the licensee's evaluation associated with the in operability of diesel generator A. This performance issue was characterized as having low to moderate risk significance ("White") in NRC Inspection Report #XXX XXXXX. During this supplemental inspection, performed in accordance with Inspection Procedure 95001, several significant deficiencies were identified with the licensee's evaluation of the inoperable diesel.

While the licensee's evaluation attributed the root cause of this issue to improper training of maintenance workers, the NRC inspectors identified that the improper maintenance was actually the result of vendor manuals that were not up to date and contained inaccurate guidance concerning the calibration of the diesel generator governor speed control unit. In addition, the inspectors determined that the vendor manual control issue does not appear to be limited to the diesel generators, as similar concerns regarding the control of vendor manuals have been documented in other NRC inspection reports. Also, the inspectors determined that the licensee's corrective actions were inadequate in that they only involved re-training the maintenance workers and failed to address the issue of vendor manual control.

As a result of these concerns, the White performance issue associated with the inoperable diesel generator will not be closed at this time. In addition, the deficiencies identified in the NRC's review of licensee's corrective actions are being considered for additional enforcement action.

Report Details

01 Inspection Scope

This supplemental inspection was performed by the NRC to assess the licensee's evaluation associated with the inoperability of diesel generator A. This performance issue was previously characterized as "White" in NRC Inspection Report #XXX XXXXX and is related to the mitigating systems cornerstone in the reactor safety strategic performance area.

02 Evaluation of Inspection Requirements

02.01 Problem Identification

a. Determination of who (i.e., licensee, self-revealing, or NRC) identified the issue and under what conditions.

The inoperability of the diesel generator was identified during a routine surveillance test performed by the licensee. During testing of diesel generator A, the diesel failed to reach the required speed, at which time the test was stopped and the diesel was declared inoperable.

b. Determination of how long the issue existed, and prior opportunities for identification

The licensee determined that the diesel was likely inoperable since last performing maintenance on September 5, 1999. The inspector agreed with the licensee's evaluation.

c. Determination of the plant-specific risk consequences (as applicable) and compliance concerns associated with the issue

The licensee's evaluation assigned a change in core damage frequency of 5 E-6 to this condition. The inspectors reviewed the licensee's evaluation and assumptions and confirmed their validity.

02.02 Root Cause and Extent of Condition Evaluation

a. Evaluation of method(s) used to identify root cause(s) and contributing cause(s).

The licensee used a combination of structured root cause analysis techniques to evaluate this issue, including barrier, change, and events and causal factor analysis. The inspectors determined that the licensee followed its procedural guidance for performing level 1 root cause analysis. The procedure required conducting interviews with key personnel and the preservation of evidence associated with the issue. The licensee successfully accomplished this by quarantining the diesel until formal troubleshooting controls could be established.

b. Level of detail of the root cause evaluation.

The licensee's root cause evaluation was thorough and identified the primary root cause of the performance issue to be poor control of vendor manuals, which resulted in the maintenance workers mis-calibrating the governor speed control unit. Furthermore, the licensee identified that the vendor manual control issue was not limited to the diesel generator but was applicable to several pieces of risk-significant equipment.

Or

The inspectors determined the root cause evaluation was not conducted to a sufficient level of detail. Although the licensee correctly diagnosed the apparent cause of the diesel failure as being a mis-adjusted governor speed control unit, the licensee's evaluation incorrectly identified the root cause as being maintenance worker error. The inspectors determined that the worker errors were actually caused by out-of-date vendor manuals for the governor speed control units. The calibration procedure in the vendor manual was for an old speed control unit that had been replaced 2 years ago. In addition, the inspectors noted that problems with control of vendor manuals for other equipment had previously been documented during NRC inspections (see NRC Inspection Reports 50-xxx/99-08 and 50-xxx/2000-05); however, the licensee had failed to enter the concerns into their corrective action program.

c. Consideration of prior occurrences of the problem and knowledge of prior operating experience.

The licensee's evaluation included a review to see if similar problems had previously been reported with the diesel governor unit. This was the first known instance of a failure of this type. The inspectors did not posses any information to the contrary.

d. Consideration of potential common cause(s) and extent of condition of the problem

The licensee's evaluation considered the potential for common cause and extent of condition associated with the lack of vendor manual control. The licensee determined that the issue of vendor manual control was not limited to the diesel generators and potentially affected other safety equipment. The inspectors agreed that this problem was not limited to the diesels, as they had previously identified problems with vendor manual control when reviewing maintenance on the auxiliary feedwater pumps. These concerns were previously documented in NRC Inspection Report 50/XXX/2000-08.

02.03 Corrective Actions

a. Appropriateness of corrective action(s)

The licensee took immediate corrective actions to make the diesel generator operable. The governor control unit was re-calibrated and the diesel generator vendor was contacted to ensure that the latest technical information was available and being used. The licensee has also specified corrective actions to address the root cause of poor vendor manual control. The licensee has begun a program to re-verify that all safety significant vendor information is current, and is planning to contact each of the associated vendors. The inspectors determined that the proposed corrective actions are appropriate.

b. Prioritization of corrective actions

The licensee's immediate corrective actions restored the diesel generators to operability within the technical specification (TS) allowed outage time. After restoring the affected diesel, the other diesel was tested to ensure that it would perform its intended functions if called upon. The inspectors witnessed this testing and observed that the diesel successfully passed the surveillance test.

c. Establishment of schedule for implementing and completing the corrective actions

The licensee's plans for the re-verification of vendor information are being implemented according to the risk significance of the equipment. The inspectors reviewed the licensee's plans for accomplishing this activity and noted that the risk significance of the equipment was being appropriately considered.

d. Establishment of quantitative or qualitative measures of success for determining the effectiveness of the corrective actions to prevent recurrence.

The licensee has enhanced its monitoring of the diesel generators to ensure that any additional failures are given appropriate management attention. The licensee has also scheduled a quality assurance audit to assess the adequacy of the corrective actions associated with the vendor manual control issue.

03. Management Meetings

Exit Meeting Summary Provide summary of exit meeting.

ATTACHMENT

Persons Contacted

Documents Reviewed (optional if list is publically available some other way)

Acronyms Used (optional)

Appendix D : Guidance For Documenting Inspection Procedure 71152 Identification and Resolution of Problems

As one of the objectives of Inspection Procedure 71152 is to provide an assessment of the effectiveness of the licensee's Problem Identification and Resolution (PI & R) programs, the type of documentation for this inspection should be different than for other baseline inspections and may include more qualitative observations. Listed below are some general principles applicable to documenting the results of IP 71152 that supplement the guidance contained elsewhere in this inspection manual chapter.

- The cover letter for this report should conform to the guidance given for other baseline inspections, but it should also contain a brief description of the team's overall conclusion regarding the effectiveness of the licensee's PI & R programs. An example cover letter is provided in the sample inspection report contained in this appendix.
- The summary of findings for this report should contain the team's overall assessment of the licensee's PI & R program based upon both the annual and the routine baseline inspections. This overall assessment should also be placed in the PIM.
- The inspection report should contain an assessment for each of the inspection requirements, as indicated in the attached example report and outline.
- Negative conclusions regarding aspects of the PI & R program should be supported by examples of performance deficiencies. Other conclusions should be supported by a brief statement of the basis of the conclusion, including the scope of material that was reviewed.

Example Inspection Report Excerpts and Outline

July 7, 2000

Mr. Charles Smith

Site Vice President

Iownanuke Power Authority

Iownanuke Unit 1

124 Atomic Blvd.

Hometown, USA

SUBJECT: IOWNANUKE UNIT 1--NRC INSPECTION REPORT NO. 50-999/00-003

Dear Mr. Smith:

On June 9, 2000, the NRC completed a team inspection at the Iownanuke Unit 1 Nuclear Power Plant. The enclosed report documents the inspection findings which were discussed on June 9, 2000, with Ms. Mary Atom and other members of your staff.

This inspection was an examination of activities conducted under your license as they relate to the identification and resolution of problems, and compliance with the Commission's rules and regulations and the conditions of your operating license. Within these areas, the inspection involved selected examination of procedures and representative records, observations of activities, and interviews with personnel.

If no findings were identified use the following:

On the basis of the sample selected for review, there were no findings of significance identified during this inspection. The team concluded that problems were properly identified, evaluated and resolved within the problem identification and resolution programs. However, during the inspection, several examples of minor problems were identified that included conditions adverse to quality that were not being entered in to the corrective action program, narrowly focused condition report evaluations, and corrective actions that were ineffectively tracked or had not occurred.

If one or more findings were identified use the following:

On the basis of the sample selected for review, the team concluded that in general, problems were properly identified, evaluated, and corrected, There was one Green finding identified during this inspection associated with the depth and effectiveness of one root cause analysis. [add one or two sentences to provide detail for each finding]. This finding was determined to be a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating this issue as a Non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this Non-cited violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region ___; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Dirojac facility.

In addition, several examples of minor problems were identified that included conditions adverse to quality that were not being entered in to the corrective action program, narrowly focused condition report evaluations, and corrective actions that were ineffectively tracked or had not occurred.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web-site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

Summary of Findings Adams Template:

IR 05000999-00-03, on 06/01-06/9/2000, Iownanuke Power Authority. Iownanuke Unit 1, annual baseline inspection of the identification and resolution of problems. A violation was identified with the licensee's root cause evaluation.

The inspection was conducted by a regional projects inspector, resident inspectors, and a regional radiation specialist. One Green issue of very low safety significance was identified during this inspection and was classified as a Non-cited violation, The issue was evaluated using the significance determination process.

Identification and Resolution of Problems

The team identified that the licensee was effective at identifying problems and putting them into the corrective action program. The licensee's effectiveness at problem identification was evidenced by the relatively few deficiencies identified by external organizations (including the NRC) that had not been previously identified by the licensee, during the review period. The licensee effectively used risk in prioritizing the extent to which individual problems would be evaluated and in establishing schedules for implementation of corrective actions. However, of the 10 root cause evaluations reviewed, one was found to be deficient in that it was not performed to a sufficient depth to determine the primary root causes of the issue. Corrective actions, when specified, were generally implemented in a timely manner. Licensee audits and assessments were found to be effective and highlighted a similar concern in the root cause area. Based on the interviews conducted during this inspection, workers at the site felt free to input safety issues into the PI&R program.

Cornerstone: Mitigating Systems

- Green. A Non-Cited Violation was identified because a deficiency was identified with the licensee's root cause evaluation RC-001 of an inoperable turbine-driven auxiliary feedwater pump. The licensee's evaluation attributed the root cause of this issue to be an improper overspeed trip setpoint caused by improper training of maintenance workers. During the inspection, NRC inspectors identified that the improper setpoint was actually the result of vendor manuals that were not up to date and contained inaccurate guidance concerning the calibration of the overspeed trip device.

The risk associated with the failure of the auxiliary feedwater pump had previously been determined to be of very low safety significance because of the Redundancy in the auxiliary feedwater system**Report Details**

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution

a. Effectiveness of Problem Identification

(1) Inspection Scope

Briefly describe the scope of what was looked at to determine whether the licensee is identifying problems at the proper threshold and entering them into the corrective action system. Include samples taken from the previous 12 months of routine baseline inspection reports. Also include in this Section the results of the team's review of licensee self assessments and audits. For example:

[The inspectors reviewed items selected across the seven cornerstones of safety to determine if problems were being properly identified, characterized and entered into the corrective action program for evaluation and resolution. Specifically, the inspectors selected 50 deviation & event reports (DERs) from approximately 2000 which had been issued between January 1999 and January 2000. The inspectors also reviewed several licensee audits and self-assessments, including two audits of the corrective action program. The effectiveness of the audits and assessments was evaluated by comparing the audit and assessment results against self-revealing and NRC-identified issues.

The inspectors evaluated the DERs to determine the licensee's threshold for identifying problems and entering them into the corrective action program. Also, the licensee's efforts in establishing the scope of problems were evaluated by reviewing pertinent control room logs, work requests, engineering modification packages, self assessment results, system health reports, action plans, and results from surveillance tests and preventive maintenance tasks. The DERs and other documents listed in Attachment 2 were used to facilitate the review.

The inspectors also conducted walkdowns and interviewed plant personnel to identify other processes that may exist where problems and issues could be identified. The inspectors reviewed work requests and attended the licensee's daily work control meeting to understand the interface between the corrective action program and the work control process.]

(2) Issues and Findings

Discuss issues and findings relative to the scope of the inspection and document general conclusions regarding effectiveness of problem identification. Included should be the basis for the general conclusion. The following provides an example of the minimum documentation which should be provided where no findings of significance were identified:

[The team determined that the licensee was effective at identifying problems and entering them into the corrective action system. This was evidenced by the relatively few deficiencies identified by external organizations (including the NRC) that had not been previously identified by the licensee, during the review period. Licensee audits and assessments were of good depth and identified issues similar to those that were self- revealing or raised during previous NRC inspections. Also, during this inspection there were no instances identified where conditions adverse to quality were being handled outside the corrective action program.]

b. Prioritization and Evaluation of Issues

(1) Inspection Scope

List the documents that were reviewed to determine whether the licensee is adequately prioritizing and evaluating issues. Include pertinent reference numbers (for example, NCR #s, violation #s, etc.).

(2) Issues and Findings

Discuss issues and findings relative to the effectiveness of the licensee's process for prioritizing issues, technical adequacy and depth of evaluations (including root cause analysis where appropriate), consideration of operability and REPORTABILITY requirements, and identification of pertinent corrective actions. Include in this Section any issues associated with the licensee's use of risk in prioritizing or evaluating issues. Document general conclusions regarding the above review,

c. Effectiveness of Corrective Actions

(1) Inspection Scope

List the documents that were reviewed to determine the timeliness and effectiveness of corrective actions. Include pertinent reference numbers (for example, NCR #s, violation #s, etc.).

(2) Issues and findings

Discuss findings and issues relative to the subject area, including the effectiveness of corrective actions to prevent recurrence. Included within this Section of the report should be an assessment of the licensee's use of risk insights in prioritizing corrective actions. Document general conclusions relative to subject area.

d. Assessment of Safety-Conscious Work Environment

(1) Inspection Scope

Describe what actions were taken to assess this subject area.

(2) Issues and findings

This portion of the report should be more general in nature, as the procedure does not contain any specific inspection requirements with regard to this subject area. Discuss findings and issues relative to the subject area. Document general conclusions relative to the subject area.

Attachments:

LIST OF PERSONS CONTACTED

<u>LIST OF DOCUMENTS REVIEWED</u> (optional if documents are identified in the body of the report)

END

ATTACHMENT 8

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

October 4, 2000

EA-00-208

Garry L. Randolph, Vice President and
Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, Missouri 65251

SUBJECT: CALLAWAY PLANT -- NRC INSPECTION REPORT NO. 50-483/00-17

Dear Mr. Randolph:

On August 11, 2000, the NRC completed an inspection at your Callaway Plant. The purpose of the inspection was to review your ALARA planning and controls. The enclosed report presents the results of that inspection which were discussed with you and members of your staff at the end of the inspection and with Mr. Ron Affolter and others by telephone on September 5, 2000.

This inspection was an examination of activities as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

This report discusses issues of low to moderate safety significance. The issues involved the failure to maintain radiation doses as low as reasonably achievable, which constitutes an apparent violation of 10 CFR 20.1101(b). As described in Section 2OS2 of this report, six jobs that accrued more than 5 person-rems each during Refueling Outage 10 exceeded their projected job doses by more than 50 percent because of a number of performance problems. This apparent violation was assessed using the Occupational Radiation Safety Significance Determination Process and was found to consist of three apparent findings, each preliminarily determined to be white. White issues have some increased importance to safety and may require additional NRC inspection. These issues have low to moderate safety significance because your 3-year rolling average, collective dose was greater than 135 person-rems for the period 1997 through 1999, which is indicative of a continuing problem with radiation dose control.

You provided your position on the preliminary inspection findings in a letter dated August 21, 2000, (ULNRC-4298) and while we believe that we have sufficient information to make our final significance determination for these preliminary inspection findings and the associated apparent violation, we are giving you the opportunity to provide us additional information on the apparent violation's significance, either in writing or at a regulatory conference. If you choose to provide additional information in writing, you should do so within 30 days of the date of this letter. Please contact Ms. Gail Good at (817) 860-8215 as soon as possible, but within 7 days of the date of this letter, to notify us of your intent. If we have not heard from you within the time specified, excepting a granted extension, we will continue with our significance determination and enforcement decision and you will be advised by separate correspondence of the results of our deliberations on this matter.

Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued for these inspection findings at this time. In addition, please be advised that the characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review. If the NRC concludes that a violation occurred, the violation will be treated in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. The current Enforcement Policy can be found on the NRC's website at www.nrc.gov/OE.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Arthur T. Howell III, Director
Division of Reactor Safety

Docket No.: 50-483
License No.: NPF-30

Enclosure:
NRC Inspection Report 50-483/2000-17

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CWY Site Secretary **(DVY)**

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket No(s).:	50-483
License No(s).:	NPF-30
Licensee:	Union Electric Company
Facility:	Callaway Plant
Report No:	2000-17
Location:	Junction Highway CC and Highway O Fulton, Missouri
Date(s):	August 7-11, 2000
Inspector:	Larry Ricketson, P.E., Senior Health Physicist
Approved by:	Gail M. Good, Chief, Plant Support Branch Division of Reactor Safety

ATTACHMENTS:

Attachment 1:	Supplemental Information
Attachment 2:	NRC's Revised Reactor Oversight Process

SUMMARY OF FINDINGS

Callaway Plant
NRC Inspection Report No. 50- 483/2000-17

IR 05000483-00-17; on 08/07-08/11/2000; Union Electric Co.; Callaway Plant. Occupational Radiation Safety Report; ALARA planning and controls.

This report documents an inspection of ALARA planning and controls conducted by a regional specialist. The significance of issues is indicated by its color (green, white, yellow, red) and was determined by the Significance Determination Process in Inspection Manual Chapter 0609.

Cornerstone: Occupational Radiation Safety

- TBD. Because of poor planning and preparation, as well as other causes, six jobs that accrued more than 5 person-rems each during Refueling Outage 10 exceeded their projected job doses by more than 50 percent. The licensee scheduled outage activities to reduce the outage duration rather than to reduce dose, failed to properly train workers in dose reduction methods, and failed to ensure good communications between radiation protection personnel and other work groups. Because of these performance problems and the licensee's history of high collective radiation doses, the NRC identified the issue as an apparent violation of 10 CFR 20.1101(b), which requires that the licensee use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable.

  Using the Occupational Radiation Safety Significance Determination Process, the NRC preliminarily determined that the violation was composed of three parts, each of low to moderate risk significance (white). Of the six jobs that exceeded their dose projections by more than 50 percent, two jobs accrued actual doses greater than 25 person-rems. Thus, because the licensee's 3-year rolling average, collective dose exceeded 135 person-rems (but did not exceed 340 person-rems) each was an apparent white finding. In addition, since there were more than two other jobs that accrued more than 5 person-rems (but less than 25 person-rems), these constituted an additional apparent white finding, for a total of three apparent white findings.

Report Details

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls 71121.02

a. Inspection Scope

The inspector interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs throughout the radiological controlled area during Refueling Outage 10. Independent radiation surveys of selected work areas within the radiological controlled area were performed. No work with potentially high exposure was conducted during the inspection. The following items were reviewed and compared with regulatory requirements:

- ALARA program procedures
- Processes used to estimate and track exposures
- Plant collective exposure history for the past 3 years, current exposure trends, and 3-year rolling average, collective dose information
- Six radiation work permit packages from Refueling Outage 10 which resulted in the highest personnel collective exposures during the inspection period
- Use of engineering controls to achieve dose reductions
- Individual exposures of selected work groups
- Hot spot tracking and reduction program
- Plant related source term data, including source term control strategy
- Radiological work planning
- ALARA-related items in Audit Report AP00-02
- Selected corrective action documentation involving higher than planned exposures and radiation worker practice deficiencies since the last inspection in this area
- Declared pregnant worker dose monitoring controls

Additionally, the criteria in NRC Manual Chapter 0610*, "Reactor Inspection Reports," Appendix E, Group 2 Questions, were used to determine whether a potential ALARA finding affected the Occupational Radiation Safety Cornerstone and whether the finding should be analyzed by the Occupational Radiation Safety Significance Determination Process. The cornerstone was affected if:

- The actual job dose associated with the finding exceeded the projected dose by greater than 50 percent;

- The licensee's 3-year rolling average, collective dose exceeded 135 person-rems/unit (for a pressurized water reactor); and

- The actual job dose associated with the finding exceeded 5 person-rems.

b. Findings

The inspector found that doses for some jobs conducted during Refueling Outage 10 were not maintained as low as was reasonably achievable. From the licensee's Refuel 10 ALARA Outage Report, the inspector determined that some jobs exceeded their dose projections by more than 50 percent and exceeded 5 person-rems per job. The following examples were noted:

Job	Radiation Work Permit	Estimated Dose (Rems)	Actual Dose (Rems)
Scaffolding in the reactor building	99-50903	22.000	46.345
Remove and install steam generator manway covers and inserts	99-53321	3.992	8.543
Eddy current/robotic plugging/stabilizing/ electrosleeving	99-53323	21.185	57.659
Health physics support for primary and secondary steam generator activities	99-53324	2.463	5.641
Foreign object search and retrieval	99-53022	1.500	6.388
Reactor coolant pump seal removal and replacement	99-52520	6.605	12.869

* 1WA

An axial offset anomaly contributed to higher than projected outage dose rates. Axial offset is a measure of the difference between power in the upper and lower portions of the core. The cause of the axial offset anomaly has been attributed to a crud buildup on fuel assemblies in the upper portion of the reactor core. (See NRC Inspection Report 50-483/97-19.) A chemical, thermal, or hydraulic shock can drive radioactive crud from the reactor core and allow it to be transported throughout the reactor coolant system where it may plate out and raise dose rates in surrounding areas.

However, the licensee acknowledged that this factor was responsible for only approximately 25 percent of the dose overrun. The licensee conducted post job reviews and identified additional causes for higher-than-projected doses. Some of the causes were common to more than one job. The inspector reviewed the post job reviews, received additional explanation of the licensee's findings from the ALARA supervisor, and reached the following conclusions:

- Some activities were not scheduled or sequenced optimally to reduce personnel dose. In an effort to advance the outage schedule, steam generator work was started three to four days earlier than normal, providing less time for radioactive decay. The licensee set up platforms around the steam generators while reactor coolant system cleanup was still in progress and before steam generator bowl drains were flushed. This also contributed to higher dose rates (Radiation Work Permits 99-53321, 99-53323, and 99-53324).

- In the original outage schedule, all reactor coolant pump seal work was to occur when the steam generator secondary sides were full. However, because all four seals had to be worked, this was not possible. To support the revised schedule, some seal work was continued with the generators empty. In past outages when this work was conducted, "an orderly proces" was followed by moving from pump to pump. This process resulted in lower personnel dose by minimizing tool movement. In Refueling Outage 10, work crews moved from pump to pump as the other work allowed. This forced the crews to move their tooling multiple times (Radiation Work Permit 99-52520).

- Insufficient mockup training was conducted to familiarize the workers with plant equipment, use of tools, and techniques to reduce dose. Workers spent more than the expected staff-hours in high dose areas because "the crews were inexperienced" and "used poor ALARA practices." Additional mockup training should have been provided to individuals that installed and removed steam generator manways and inserts and those that used robotic eddy current equipment (Radiation Work Permits 99-53321, 99-53323, and 99-53324).

- Communication between radiation protection personnel and primary contractor personnel was "poor." Radiation protection personnel "seldom" knew job status or the schedule for the upcoming shift. Therefore, they could not plan their activities to reduce dose (Radiation Work Permits 99-53324 and 99-53022).

- There was a "lack of involvement and ownership" of the scaffolding program by craft supervisors. Reviews of scaffolding packages were not completed in a timely manner. Alternatives to scaffolding erection were not pursued. Scaffolding was allowed to be erected during times in the outage when dose rates were high, such as during reactor coolant system cleanup (Radiation Work Permit 99-50903).

The inspector also found that high collective radiation dose has been a continuing problem. The licensee's 3-year rolling average, collective dose exceeded 135 person-rems in 1999 and increased from 1997 through 1999. Dose information obtained from the licensee is shown in the following chart.

	1996	1997	1998	1999
Annual Collective Dose	248	12.5	200.7	320
Outage Dose	232	NA	185	305
	1994-1996	1995-1997	1996-1998	1997-1999
3-Year Average Collective Dose	149.8	149.2	153.7	177.7

The inspector determined through conversations with members of the Office of Nuclear Reactor Regulation that the licensee's 3-year rolling average, collective dose for 1997 through 1999 was the second highest among pressurized water reactors. This will be documented in NUREG 0713, Volume 21, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities 1999."

10 CFR 20.1101(b) requires that the licensee use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA). Because the licensee had a history of high collective doses, scheduled outage activities to reduce the outage duration rather than to reduce dose, failed to properly train workers in dose reduction methods and failed to ensure good communications between radiation protection personnel and other work groups, the inspector identified the failure to maintain doses resulting from six Refueling Outage 10 jobs as low as was reasonably achievable as an apparent violation of 10 CFR 20.1101(b). Specifically, it appears that the licensee did not use, to the extent practical, procedures and engineering controls based on sound radiation protection principles to achieve occupational doses ALARA. This finding is in the licensee's corrective action program as Suggestion Occurrence Solution 00-0377 (50-483/0017-01).

The inspector used the Occupational Radiation Safety Significance Determination Process and preliminarily determined that the violation was composed of three parts, each of low to moderate risk significance (white). Of the six jobs that exceeded their dose projections by more than 50 percent, two jobs accrued actual doses greater than 25 person-rems. Thus, because the licensee's 3-year rolling average, collective dose exceeded 135 person-rems (but did not exceed 340 person-rems) each was an apparent white finding. In addition, since there were more than two other jobs that accrued more than 5 person-rems (but less than 25 person-rems), these constituted an additional apparent white finding, for a total of three apparent white findings.

4. OTHER ACTIVITIES

4OA6 Management Meetings

.1 Exit Meeting Summary

The inspector presented the inspection results to Mr. G. Randolph, Vice President and Chief Nuclear Officer, and other members of licensee management at the conclusion of the inspection on August 11, 2000. The licensee disagreed with the potential significance of the findings presented and submitted its position to the NRC in a letter dated August 21, 2000 (ULNRC-4298).

The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

During a telephone conference on September 5, 2000, the inspector informed Mr. R. Affolter and other members of the licensee staff that the findings were an apparent violation of 10 CFR 20.1101(b).

ATTACHMENT 1

Supplemental Information

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Affolter, Plant Manager
R. Farnam, Supervisor, Health Physics Operations
K. Gilliam, Supervisor, Radiation Protection and Chemistry
J. Hiller, Engineer, Quality Assurance
J. Laux, Manager, Quality Assurance
G. Randolph, Vice President and Chief Nuclear Officer
M. Reidmeyer, Supervisor, Regional Regulatory Affairs
R. Roselius, Superintendent, Radiation Protection and Chemistry
W. Witt, Assistant Plant Manager

NRC

B. Baca, Health Physicist
J. Hanna, Resident Inspector
M. Shannon, Senior Health Physicist

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-483/0017-01	AV	Failure to maintain radiation doses as low as is reasonably achievable (Section 2OS2)

Opened and Closed During this Inspection

None

Previous Items Closed

None

Previous Items Discussed

None

DOCUMENTS REVIEWED

Refuel 10 ALARA Outage Report

RCS Shutdown and Startup Evaluation for Refuel 8

Audit Report AP00-02

APA-ZZ-01000, “Callaway Plant Health Physics Program,” Revision 15

APA-ZZ-01001, “Callaway Plant ALARA Program,” Revision 6

APA-ZZ-01102, “Pre-Job ALARA Planning and Briefing,” Revision 15

HTP-ZZ-01103, “Post-Job ALARA Review,” Revision 12

HTP-ZZ-01201, “Preparation and Maintenance of General and Specific Radiation Work Permits,” Revision 30

HTP-ZZ-01203, “RWP Access Control,” Revision 27

Supplemental Information - Inspection Report No. 50-483/2000-012 (ULNRC-4298) dated August 21, 2000 (the report number changed due to the need to issue a stand-alone report).

ATTACHMENT 2

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the significance determination process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, or RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an action matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the action matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.