January 26, 2005

NRC REPORT TO IEEE/NPEC

1.0 Key Topics

1.1 License Renewal

Updating of License Renewal Guidance Documents (LRGDs)

The LRGDs consist of GALL, SRP-LR (NUREG-1800) , RG 1.188, and NEI 95-10. These guidance documents were developed for the license renewal (LR) review process to be more effective, efficient, and have been used for LR since 2001. Based on the experience gained from LR review process, the applicants and the staff have recognized a need for upgrading the documents to address newly emerged technical issues, and revisions to various aging management programs (AMPs). For this effort, the industry has been very responsive by standardizing their LR application format to gain further efficiency and to reduce the LR review period. The updating process requires the public and industry participation and preparing the supportingbasis documents for the changes. The staff has posted preliminary draft updates to GALL and SRP-LR (NUREG-1800) on September 30, 2004 on its website. This also includes a draft document that provides explanation and justification for the updates. A public meeting was held in November 2004 and the staff received positive feedback from this meeting. A meeting to finalize NEI 95-10 (rev. 5) was also held on January 13, 2005. Draft updates to GALL and SRP-LR will be available for public comment by January 31, 2005 (comment period will expire on March 30, 2005). The guidance documents will be finalized by September 2005.

Program reforms and updated guidance documents will enhance the efficiency and reduce costs of license renewal program. Increased use of site audits will increase effectiveness and efficiency, reduce scope of staff review, and increase use of contractors, which will leave more time for staff to attend to complex policy work.

Renewal applications reviews have been completed for 26 reactors at 15 sites. NRC expects renewal applications for eight units in fiscal 2005 and seven units in FY 2006.

Status of License Renewal (attachment 1-routed).

1.2 10 CFR 50.69

Federal Register Notice dated November 22, 2004; PP 68007
The Commission is establishing Sec. 50.69 as an alternative set of requirements whereby a licensee or applicant may undertake categorization of its SSCs consistent with the requirements in Sec. 50.69(c) and adjust treatment requirements per Sec. 50.69(d) based upon the resulting significance. Under this approach, a licensee or applicant is allowed to remove the special treatment requirements listed in Sec. 50.69(b) for SSCs that are determined to be of low safety significance while potentially enhancing requirements for treatment of other

SSCs that are found to be safety significant. The requirements establish a process by which a licensee categorizes SSCs using a risk-informed process, adjusts treatment requirements consistent with the relative significance of the SSC, and manages the process over the lifetime of the plant. To implement these requirements, a risk-informed categorization process is employed to determine the safety significance of SSCs and place the SSCs into one of four Risk-Informed Safety Class (RISC) categories. It is important that this categorization process be robust to enable the Commission to remove requirements for SSCs determined to be of low safety significance. The determination of safety significance is performed by an integrated decision-making process which uses both risk insights and traditional engineering insights. The safety functions include both the design basis functions (derived from the ``safety-related'' definition, which includes external events), as well as functions credited for severe accidents (including external events). Treatment requirements for the SSCs are applied as necessary to maintain functionality and reliability and are a function of the category into which the SSC is categorized. Finally, assessment activities are conducted to make adjustments to the categorization and treatment processes as needed so that SSCs continue to meet applicable requirements. The rule also contains requirements for obtaining NRC approval of the categorization process and for maintaining plant records and reports.
Regulatory Guide 1.201 will be revised to implement this rule.

1.3 Fire Protection Rulemaking 10 CFR 50.48:

A Duke Energy plant is expected to be the first one to transition to the new risk-informed fire protection regulation adopted in July 2004. The new regulation allows a licensee to comply with the NFPA Std 805, with certain exceptions, as an alternative to existing fire protection requirements.

1.4 Risk-Informed Regulation Implementation Plan:

The staff provided to the Commission an update of the risk-informed regulation implementation plan (see SECY-04-197 dated October 25, 2004) (attachment 2-routed).

1.5 Power Uprates

See attachment 3 (routed) for a listing of Approved Applications for Power Uprates.
The latest survey of planned uprates indicates that licensees plan to submit over the next five years uprate applications for 26 units. As a result of these uprates, electric generating capacity could increase by 1,766 MW.

1.6 Draft Report on," Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1986-2003"

In December 2004, NRC issued a draft report titled, "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1986-2003." This report is an update of two previous analyses of loss of offsite power (LOOP) event at U. S. Commercial nuclear power plants. Frequency and duration estimates for critical plant

operation and shutdown operations were generated for five categories of LOOPs (plant centered, switchyard centered, grid related, severe weather related, and extreme weather related). Overall findings are LOOP frequencies have significantly decreased in recent years during plant operation, while LOOP durations have increased. This information is needed for probabilistic risk assessment models of U.S. plants to accurately model current risk from LOOP and associated station blackout scenarios.

LOOP is a loss of offsite power to all safety buses, while a station blackout is the loss of all offsite and onsite ac power to the safety buses.

1.7 St. Lucie Breaker Failures

NRC inspectors recently finished up the second phase of a two-part inspection into the failures of two 4.16 KV safety-related breakers at St. Lucie. The first malfunction occurred on Dec 9, 2004, when a component cooling water pump breaker did not close as expected during a post-maintenance test. The second malfunction, on Dec. 13, involved a separate component cooling water breaker that failed on a test stand during preventive maintenance. Both breakers were Westinghouse DHP magnetic air circuit breakers. The licensees plan to replace all 60 circuit breakers.

1.8 Set Point Calculation: Method 3

The staff has questioned about the adequacy of a methodology used to calculate settings for instruments that monitor operational safety limits at nuclear power plants. At issue is "Method 3" as defined in ISA S67.04, Part II (1994). About 70% of power reactor licensees use Method 3. The Allowable values (AVs) calculated using Method 3 may not provide adequate margin to assure that the analytical limit is not violated. AV must ensure that the probability of trip at Analytical Limit (AL) is never less than 95% but with Method 3 failure to trip is always more than 5%. The details of the dispute with the nuclear industry over Method 3 are technically complex.

2.0 Research Activities:

2.1 Update of the NRC Digital Systems Safety Research Plan

The staff is in the process of updating its Digital Systems Safety Research Program Plan. The last revision (SECY-01-0155), covered research planned for FY 01-04, and the new plan will cover research planned for FY 05- 09. As part of the development of the research plan, the NRC is planning to interact with the interested stakeholders.

2.2 Research Project, "Effects of Switchgear Aging on Energetic Faults"

An event at Taiwan nuclear power plant resulted an energetic fault and left with a fire that damaged medium voltage (4.16 kV) switchgear. The damage affected both safety division buses in the plant and the event could have resulted in a station blackout if an emergency diesel was not recovered on time. Since the design of switchgear was based on the US specification, such energetic fault can occur in US nuclear plants. The research identified that aging of safety-related

power system components could contribute and increase frequency and severity of such energetic faults if it is not properly managed. The NRC initiated a research project to evaluate the effects of switchgear aging on energetic faults and to perform the sensitivity analysis. A system level evaluation of medium voltage electrical distribution systems was completed on December 31, 2004 to exam the effects of various system transients. This task involved a review of plant electrical distribution system designs on transients. Two representative nuclear plant distribution systems were modeled using power system analysis software (ETAP) to aid the evaluation of medium voltage system operating and fault transients. Impact of equipment aging on system protection will be reviewed. A final NUREG/CR is expected to be issued by the end of this year.

2.3 Risk Information for licensing digital I&C

The NRC is conducting a set of research programs to develop needed information, and methods to support the use of risk information in the licencing of digital instrumentation and control systems used in nuclear power plants. This research is expected to produce a draft regulatory guide to support risk informed digital systems reviews within the next year and the supporting information in NUREG/CRs.

2.4 International Workshop on International Workshop on Systems Software Engineering

During the week of September 6-10, 2004, an International Workshop was held in Istanbul, Turkey on Software Systems. This was co-sponsored by the Nuclear Energy Agency of the Organization for Economic Cooperation and Development, the International Federation of Automatic Control, the Center for Technology Risk Studies of University of Maryland, and the NRC. The workshop brought together the experts in three aspects of software based systems development: system engineering, software engineering, and real-time programming. This workshop provided the participants with insights as to how to improve methods for developing and modeling software, specifically in the areas of software validation and verification, and to support improving the development of analytical software and the assessment of real-time software-based systems..

At this Workshop, there were several discussion on the issue of software QA (SQA) for both high reliability real time systems and analytical systems. IEEE Std 1012 and Std 1028 as well as IEC standards are becoming more and more popular in the area of SQA of non real time software used to support safety decisions.

2.5 ANS Topical Meeting on I&C and Human Machine Interface Technologies

During September 19-22, 2004, the ANS International Topical Meeting on Nuclear Plant Instrumentation, Control, and Human Machines Interface Technologies was held in Columbus, Ohio.

3.0 NRC Rulemaking and Regulatory Guide Activities:

3.1 Draft Regulatory Guide DG-1130, “Criteria for Use of Computers in Safety Systems of Nuclear Power Plants,” was issued in December for public comment.

This draft Regulatory Guides provides guidance for the use of computers in safety systems and includes cybersecurity guidance.
The comment period will expire on February 11, 2005.

3.2 Draft Regulatory Guide DG-1139, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," was issued for public comment in September 2004. The comment period expired on December 15, 2004.

The staff received 3 comment letters (including from NEI & STAR). A summary of the significant comments on the draft Reg Guide is as follows:

1. The industry questioned the requirement for a risk impact evaluation for plant changes to deterministic requirements, claiming that NFPA 805 is contradictory on this issue.

2. The industry appealed to the NRC to provide an explicit SER when approving the license amendment request to clearly establish the elements of the new licensing basis to minimize questions of compliance during inspections. They also recommended that a detailed transition summary document that explicitly addresses every major area of fire protection be submitted to and approved by the NRC.

3. The industry requested more details on the approach to transitioning GL 86-10 evaluations from the current licensing basis to the new 805 licensing basis.

4. The industry requested that the Regulatory Guide endorse the various NEI industry guidance documents currently with the NRC for review.

5. The industry believes that tracking small changes in risk is an undue burden that does not result in any benefit or improved nuclear safety.

6. The industry requested more specifics on acceptance criteria for defense-in-depth, safety margins, spurious actuations and recovery actions (operator manual actions).

7. The industry requested more guidance on the use and acceptance of alternative methods to those specified in NFPA 805.

3.3 Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants according to their Safety Significance," was recently issued on trial basis (DG-1121).

3.4 Draft Regulatory Guide DG -1137 " Guidelines for Lightning Protection in NPPs" expected to be published by December 2005.

3.5 Draft Regulatory Guide DG-1128, which is a revision of Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," is under development. This revision will endorse IEEE Std 497-2002.

3.6 Regulatory Guide 1.75, “Criteria for Independence of Electrical Safety Systems,” is expected to be issued in February 2005. This guide endorses IEEE Std 384-1992 with exceptions.

4.0 NUREGs

4.1 NUREG/CR-6848, “Preliminary Validation of a Methodology for Assessing Software Quality”, July 2004.

4.2 NUREG/CR-xxxx, " A Technical Basis for Regulatory Guidance on Lightning Protection in NPPs," publication expected in December 2005 .

4.3 NUREG/CR-xxxx, "Preliminary Assessment of Wireless Technologies and Their Application to NPPs," publication expected in May 2005.

4.4 Draft NUREG-1791 “Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)” was issued for public comment in September 2004. Final is expected in Summer of 2005.

4.5 Draft NUREG-1792 was issued for comment on good practices for Human Reliability Analysis when performing PRAs.

5.0 Human Factors

There is an increasing interest in Human Factors work associated with advanced and evolutionary reactors in the international community.

Safety Culture and Safety Management are the focus of several countries.

Spain is adapting the Reactor Oversight Process (ROP) to their inspection program, but expect to include PIs for cross-cutting issues.

The Commission directed the staff (SRM-04-0111) to publish a Safety Conscious Work Environment (SCWE) Regulatory Information Summary to serve as a “good practices” document. It was published for public comment in the Fall of 04 and is now being prepared in final form. This SRM also directed the staff to enhance the ROP to better account for Safety Culture. Staff plans to develop proactive means (baseline inspection and objective measures) to monitor safety culture at NPPs and to develop a reactive inspection process for plants in column 3 or higher of the ROP action matrix. This will also involve training of inspectors and monitoring industry (INPO) activities.

A revision to 10 CFR Part 26, Fitness-for-Duty, is planned to be published for public comment in the summer of 2005. This revision will incorporate new guidance on working hours and fatigue management and will update the drug and alcohol provisions to be consistent with HHS guidelines.

Three new research efforts have been/will be initiated in FY 05

Indicators of Human Performance - This is a feasibility study to determine if new PIs can be developed to support the Human Performance cross-cutting issue for the Reactor Oversight process.

Latent errors - This will also be a feasibility study to determine if the inspection methods can be developed to identify latent errors, before they become a factor in an event. There is a companion human reliability analysis (HRA) study to better incorporate latent error in HRAs.

Human Factors in Advanced Reactors - The initial effort in this project is to prioritize human factors efforts as they relate to issues associated with new generation of reactors.

6.0 Recent NRC Generic Communications

6.1 Generic Letter(s):

None issued.

EXPECTED:

1. Generic Letter on Medium Voltage (4.16 kV) Underground Cable Aging Issues

NRC staff plans to issue a generic letter on potential degradation concerns identified in the medium voltage underground cables during a review of the license renewal and operating experience on emergency component cooling water system cable problems at a nuclear power plant. The staff has reviewed the operational data (LERs) and finds that medium voltage cables can fail due to water treeing or a decrease in a dielectric strength due to aging, if it exposed to condensation and wetting in inaccessible locations such as conduits, cable trenches, and duct banks. In deciding this issue should be reviewed under 10 CFR Part 50 (i.e., generic letter), the staff has issued a letter (ML040370628) titled, "Potential Common Mode Failure of Medium Voltage Underground Cables" to NEI to engage the industry regarding the potential common-mode failure of those medium voltage cables on February 5, 2004. The staff had met with NEI on June 2, 2004 to seek the clarification of the issues and for discussion of the above staff's concerns. NEI agreed that cable water trees could be a problem for those medium voltage underground cables and susceptible to degradation, but it occurs only randomly, disagreed with the staff opinion of a common mode failure implication requiring an immediate resolution.

2. Generic Letter on “Grid Reliability, Impacts on Plant Risk, and Operability of Offsite Power”

In connection with August 14, 2003, blackout event in Northeastern United States of America and a part of Canada, NRC had issued Regulatory Issue Summary (RIS) (ML040990550) on April 15, 2004 to advise nuclear power plants of the requirements of Section 50.65 of Title of the 10 CFR (10 CFR 50.65), "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," 10 CFR 50.63, "Loss of all alternating current power," 10 CFR Part 50 Appendix A, GDC 17, "Electric power systems," and plant technical specifications on operation of offsite power. Subsequent to the issuance of the RIS, the staff had issued TI (ML040360320) titled, "Offsite power system operational readiness" for these inspections on April 29, 2004. This TI was to confirm, the operational readiness of offsite power system in accordance with NRC requirements prescribed in 10 CFR Part 50, GDC-17, Plant TS for offsite power systems, 10 CFR 50.63, and 10 CFR 50.65(a)(4). After reviewing information collected via this TI, the staff determined that a generic letter is required to assess grid reliability, impacts on plant risk, and operability of offsite power.

6.2 Information Notice(s):

IN 2004-19, " Problems Associated with Back-UP Power Supplies to Emergency Response Facilities and Equipment," November 2004.

6.3 Regulatory Issue Summaries

RIS 04-003, Rev 1, "Risk-Informed Approach for Post-Fire Safe-Shutdown Circuit Inspections. (Attachment 4- routed).

7.0 Need for new standards:

A standard is needed to address the security aspects of digital systems. IEEE Std 7-4.3.2-2003 did not address this topic. However, NRC has included cyber security requirements in DG-1130, "Criteria for Use of Computers in Safety Related Systems of Nuclear Power Plants." A standard should be developed to address these security requirements.

A standard or good practice document should also be developed for root cause analysis.

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