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Chief, Rules and Directives Branch
Office of Administration
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

Subject: **Draft Regulatory Guides, DG-1114 - Control Room Habitability at Light-Water Nuclear Power Reactors, and DG-1115 - Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors**
Request for Comments

Ladies and Gentlemen:

Carolina Power & Light and Florida Power, subsidiaries of Progress Energy Company, endorse NEI's comments on the subject guidance document. In addition, we propose the two comments attached.

We encourage the NRC to work together with NEI towards the resolution of these comments and the remaining issues.

Please contact Tony Groblewski at (919) 546-4579 if you have questions.

Sincerelv.

Terry C. Morton
Manager - Performance
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RGH

Template = ADM-013

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Comments on Draft CRH Regulatory Guides

1. DG-1115, Section C.3, 1st Paragraph:

   The proposed wording places an unnecessary expense on those licensees who have already performed two or more integrated tests and have demonstrated acceptable performance of the CRE. The paragraph should be revised to allow licensees who have demonstrated insignificant degradation of the CRE based on two or more test results to be exempt from further testing until appropriate performance-based testing frequencies can be developed based on industry experience.

2. General comment that applies to DG-1113 and 1114

   Regulatory Guide 1.78, *Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release*, Revision 1 was issued in December 2001. This Regulatory Guide includes Section C.2 – Risk Evaluation. This Section states the following:

   > "Release of toxic chemicals that have the potential to result in a significant concentration in the control room need not be considered for further detailed evaluation if the releases are of low frequencies (10^{-6} per year or less) because the resultant low levels of radiological risk are considered acceptable. If demonstrated, an acceptable level of risk may be used by licensees to support license amendment requests."

   It is recommended that this same approach be applied to radiological release events. Similar statements should be added to Draft Regulatory Guide 1114, and would be applicable to toxic gas and radiological events.

   Incorporating this criterion into Draft Guide 1114 would result in the need for significant changes to Draft Regulatory Guide 1113, *Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors*, and Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*. The reason is that these Regulatory Guides (as well as the current NRC Standard Review Plan) specify radiological release events that have a probability significantly less than 10^{-6} per year.

   As an example, Attachment 1 discusses the probability of the Steam Generator Tube Rupture Event as specified by the regulatory guides. The specified design basis event requires the assumption of a SGTR occurring, coincident with operation at the Technical Specification reactor coolant activity limit and a loss of offsite power. As quantified in Attachment 1, the estimated probability of this design basis event is 10^{-8} per year. This is the probability of an 'event that has the potential for high concentrations in the control room'. The probability of actually getting such high concentrations in the control room is significantly less. The low probability event must be combined with adverse meteorological dispersion conditions and a wind direction blowing towards the control room. Even if the wind were blowing towards the control room, since the significant releases are from a very hot jet of steam directed upwards by the main steam safeties or atmospheric dump valves, the plume would likely pass well above the control room roof/intake. Hence, the most likely dose in the control room is zero. Such an extremely low probability event should not be a design basis event. Such an extremely low probability event should not dictate design considerations for the control room habitability systems. Industry resources should not be expended analyzing such a low probability event.

The Regulatory Guides for performing dose analyses need to be revised to specify design basis events that have a probability in the range of 10^{-6} per year. If a credible set of assumptions results in doses that are obviously well within dose acceptance criteria, then that accident should be eliminated from the guidance as an accident that requires a dose consequence analysis.

For example, for a SGTR event, a credible set of conditions might be a tube rupture simultaneous with high reactor coolant activity. However, the analysis would not require the assumed loss of offsite power. The conclusion of such an exercise would be that any combination of probable SGTR conditions would result in insignificant doses (a few millirem or less).

This conclusion is supported by the dose consequences from the actual steam generator tube rupture (or large leak) events that have occurred in the industry. The public dose from these events was small fractions of a millirem. The only event where the conservatively estimated dose was between 1 and 5 millirem was the 1982 event at Ginna. The higher Ginna dose was due to the fact that water was discharged from the main steam safety valves. The control room dose from these actual events was undetectable. There is no way for a SGTR event to approach the dose limits (e.g. – 30 REM thyroid) without combining many conservative but low frequency assumptions, thus bringing the probability to much less than 10^{-6} per year.

The logical, risk-based conclusion would therefore be that plant-specific dose analyses for a SGTR are not required due to the known low dose consequences of a credible event. This would be a valid conclusion for both control room habitability analyses and FSAR Chapter 15 public dose analyses. A review of the specified criteria for many of the other accidents (e.g. – Control Rod Ejection, Main Steam Line Break) would likely result in the same conclusion. This would significantly reduce the analyses requirements and NRC review requirements related to design basis dose consequences, with no increase in public risk.

As part of the defense-in-depth philosophy of nuclear power and the desire to use deterministic analyses, it is recommended that the LOCA, with the source terms as specified in either TID-14844 or Regulatory Guide 1.183, continue to be analyzed as the postulated design basis accident for 10 CFR 100 siting criteria and control room habitability design. However, risk based considerations should also be applied to the methods/assumptions which are imposed on the LOCA dose analyses. This would recognize the fact that the base probability of the event (a core melt with containment leakage at the Technical Specification limit) is approximately 10^{-6} per year. Therefore, there should be no requirement to use a conservative estimate for every other assumption.

For the LOCA core melt dose analysis, nominal or best-estimate values could be used. In the current specified approach, the analysis must assume:

- the 95% worst case meteorological dispersion instead of average dispersion,
- the minimum possible control room filter recirculation flow allowed by Technical Specification instead of measured or rated fan flows,
- 2% reactor power measurement uncertainties and end-of–core life assumptions for estimating core nuclide inventories,
- and continuous occupancy of the control room by the same individual for 24 consecutive hours following the accident.

The list of conservative assumptions continues, each assumption possible, but with a low probability. When combined, the overall probability of "the event" analyzed is orders of magnitude less than 10^{-6} per year. Specifying the use of best-estimate/nominal assumptions would also resolve issues, such as the current issue on how to address the uncertainty associated with control room envelope tracer gas in-leakage testing. For low inleakage plants, the uncertainty of the test method is often greater than the measured inleakage (e.g.- 14 cfm ± 30 cfm). A risk-based approach would allow the use of the best-estimate value (14 cfm in the example).

In response to a question at the NEI sponsored Control Room Habitability Workshop in August 2001, Mr. Mark Reinhart (Section Chief, NRC/NRR), stated that the NRC would be receptive to evaluating dose analysis methodology that use best-estimate analyses and applies an overall safety factor on the result. However, he noted this would be a future consideration. The industry would gain the most benefit if this philosophy were considered now as part of the CR Habitability Regulatory Guide reviews.

It is during the next 5 years that many licensees will be reanalyzing their design basis radiological analyses. Some because they will perform tracer gas inleakage testing and the measured results will exceed the analysis assumptions, some to change to the Alternative Source Term criteria, some to support power up-rate license amendments, and some to support license extension amendments. Taking the time in 2002 to revise Draft Guide 1113 and Regulatory Guide 1.183 to specify the credible accidents and assumptions that should be analyzed will save significant licensee and NRC resources for every future submittal. This significant savings will come with no increase to public risk and could result in design/procedure changes that improve public risk.

Attachment 1

Steam Generator Tube Rupture Example
Of a Low Probability Event

Note – the probability values listed below are rounded to the nearest factor of 10. Each could be argued to be somewhat higher or lower, however such refinements would not change the overall conclusion. A more rigorous analysis could be performed.

Note – the following addresses some of the more basic assumptions that have a significant effect on the dose. There are many other assumptions that are assumed to be at the conservative end of their probabilistic value. This includes iodine removal factors in the steam generator, primary to secondary break flow rates, an assumed single failure usually extending cooldown time, control room intake flow rates, control room recirculation flow rates, filter efficiencies, occupancy times, and breathing rates. Additionally, this analysis calculates the probability of an event that has the 'potential' to result in high concentrations in the Control Room. This potential still requires low probability conditions, such as minimal atmospheric dispersion and the wind blowing towards the control room from the release point. Hence, the actual probability of having high concentrations in the control room is much less than the low probability of the event determined below.

Probability of initiating event (tube rupture) – 10^{-3} per year

Comments – the event analyzed is a guillotine rupture of a tube, thus maximizing the flow rate compared to some of the high leak rate events observed in the industry. The rupture is also often assumed at the top of the tube sheet to minimize iodine removal in the steam generator. The probability recognizes the improvements made over the last 20 years in SG tube inspections and repairs and in leak monitoring and is approximately equal to the base probability for a SGTR in the PRA analysis.

Probability of event occurring while at maximum coolant activity – 10^{-3}

Comments – the bounding accident for control room habitability is the event that starts with a pre-accident spike, which for most licensees is 60 μCi/gm dose equivalent I-131. It is doubtful that any plant has ever experienced an iodine concentration this high. Some plants have spiked above the 1 μCi/gm dose equivalent I-131 level, but only for a few hours out of the one-year when they were operating above 10% of the Technical Specification steady state limit. With the improvements in fuel design, typical operation for nearly all PWR's is at coolant activities 100 to

1000 times less than the 1 μCi/gm dose equivalent I-131 limit. The dose consequences are directly related to the coolant activity.

Probability of a loss of condenser – 10^{-2}

Comments – The current required assumptions force the loss of the condenser, as a loss of offsite power must be assumed. Due to the improvements made in the power grid stability since the 1960's it is not expected that a plant trip caused by the SGTR will result in a loss of the offsite power supply. There have been numerous nuclear plant trips in the industry, none of them resulting in a loss of offsite power. The significance of this assumption in the dose analysis is the loss of the condenser as the steam flow path. The condenser provides an iodine removal factor of at least 1000. The probability of loss of offsite power during the few hours following the SGTR is likely less than 10^{-3}, however, since there are other ways to lose the benefit of the condenser, such as a stuck open main steam safety valve, a probability of 10^{-2} is assumed.

Combined probability of event = 10^{-3} / yr x 10^{-3} x 10^{-2} = 10^{-8} per year