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December 17, 1998

Dr. Carl A. Paperiello, Director
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

REFERENCE: 10 CFR Part 70 Nuclear Criticality Safety

Dear Dr. Paperiello:

At the December 3-4, 1998 *NRC Public Meeting on Amendment to 10 CFR Part 70* the Nuclear Energy Institute (NEI)[1] expressed concern over Nuclear Regulatory Commission (NRC) proposals addressing Nuclear Criticality Safety (NCS) in the proposed revisions to 10 CFR Part 70. This letter documents such concerns and recommends improvements to the proposed rule revisions in the area of NCS. Comments and recommendations on the rule implementation mechanisms, as detailed in the draft NUREG-1520, Standard Review Plan, will be presented by NEI and industry representatives at the *NRC Workshop on NCS* scheduled for January 13-14, 1999. This letter defers discussion of highly technical implementation issues to that Workshop and addresses only revision of the Part 70 rule language.

NEI supports the NRC's efforts to make the Part 70 rule consistent with the ANSI/ANS-8 NCS standards. In this regard, some modification of the language of the proposed revisions is, however, required to focus on the ***risks***, rather than the 'consequences' and 'quantified likelihood' of accident sequences that could lead to potential nuclear criticalities. While nuclear criticality accidents can be
considered high consequence events, their risks differ. Consequently, the safety controls required to mitigate the risk of an accident sequence and the assurances

[1] NEI is the organization responsible for establishing unified nuclear industry policy on matters affecting the nuclear energy industry, including the regulatory aspects of generic operational and technical issues. NEI's members include all utilities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect/engineering firms, fuel fabrication facilities, materials licensees, and other organizations and individuals involved in the nuclear energy industry.

to be applied to such controls should be graded according to the severity of the potential risk. Such safety controls and assurances to ensure sub-criticality in fuel cycle operations cannot be arbitrarily assigned, but must be selected (and graded) based upon the results of an Integrated Safety Analysis (ISA).

NEI believes that Part 70 revisions should impose a level of regulatory oversight over NCS similar to that afforded to other potential hazards, as NCS differs in no intrinsic way from other industrial safety concerns. Experience gained from U.S. nuclear criticality accidents indicates that a danger to health is posed only when a nuclear worker is within a few meters of a critical excursion (and with little intervening shielding). Nuclear criticality accidents pose no danger to the public.

The risk of an inadvertent nuclear criticality accident can be minimized through application of risk-informed, performance-based regulation. A Part 70 license should include license commitments to manage NCS in accordance with ANS-8 guidelines. It should define the broad, operational bases for a facility, within which limits the licensee may safely operate without additional NRC approval (or license amendment) and without burdensome reporting requirements. A licensee should have the latitude to focus its NCS resources on high-risk nuclear criticality accident sequence prevention and to address safety issues within a licensee's corrective action program.

NEI looks forward to continuing our dialogue with the NRC on the Part 70 rulemaking. We should be pleased to address any questions which you or your staff may have on the industry's concerns and positions.

Sincerely,

Marvin S. Fertel

Enclosure
cc: The Honorable Shirley A. Jackson, Chairman, NRC
The Honorable Greta J. Dicus, Commissioner, NRC
The Honorable Nils J. Diaz, Commissioner, NRC
The Honorable Edward McGaffigan, Jr., Commissioner, NRC
The Honorable Jeffrey S. Merrifield, Commissioner, NRC
Dr. William D. Travers, Executive Director for Operations, NRC

ENCLOSURE

NUCLEAR ENERGY INSTITUTE (NEI)
RECOMMENDED LANGUAGE CHANGES TO PART 70
CONCERNING NUCLEAR CRITICALITY SAFETY

I. Proposed Changes to the Draft Language

(a) Risk-Informed Regulation

Proposed revisions to 10 CFR Part 70 addressing Nuclear Criticality Safety (NCS) are ambiguous and could potentially be misinterpreted as 'consequence-based' rather than 'risk-based' regulation. Proposed revisions continue to address the consequences and likelihood of an accident sequence, whereas they should focus regulatory attention on its ***risk***. The language of the proposed revisions should be checked to ensure consistency with the concept of *risk*.

The fuel cycle industry acknowledges that a nuclear criticality accident is an operating hazard whose risk must be adequately managed. The operating history of fuel cycle facilities has shown that serious risks to worker health and safety from a nuclear criticality accident are confined to within several meters of the critical excursion. Most criticality accidents in the U.S.A. have not exposed workers to greater than 100 rem. Never has the health of a member of the public been impacted by a nuclear criticality accident in the U.S.A. Risks to workers from other operating hazards such as fires, equipment malfunctions or hazardous chemical releases often pose greater health and safety concerns. In addition, some high consequence accident scenarios may be low risk due to their inherently low likelihood of occurrence (e.g. certain natural phenomena). Depending on the structures, systems and controls implemented at a facility, the *risk* of a nuclear criticality accident – the parameter to be addressed by the NRC in risk-informed, performance-based regulation – may indeed be low.

NCS revisions to Part 70 should consider application of a risk-informed, performance-based methodology to:

- evaluate the *risk* (i.e. consequences <u>and</u> likelihood) of potential nuclear criticality accidents whether initiated by external events, process deviations or internal events
- establish appropriate *risk-based* (graded) levels of protection to prevent nuclear criticality accidents
- establish appropriate *risk-based* (graded) levels of assurance for items relied on for safety to ensure their availability and reliability

(b) *Double Contingency*

Draft revisions to 10 CFR Part 70 define, and now require, double contingency as a practice for managing the risk of a nuclear criticality accident. The fuel cycle industry has for many years adhered to American National Standard ANSI/ANS-8.1 and incorporated the double contingency principle into its NCS programs. The NRC has consistently endorsed the ANSI/ANS-8.1 double contingency principle and now does so in the draft revisions to Part 70.

NEI is concerned with the manner in which NUREG-1520, Standard Review Plan (SRP) for review of Part 70 license applications, requires implementation of the double contingency principle. To determine whether there are at least two 'unlikely', independent and concurrent process changes necessary before a criticality might occur (i.e. double contingency protection), industry has relied on the expertise, experience and judgment of nuclear criticality experts on a deterministic basis. Risk-informed decisions are reached by thoroughly understanding the system characteristics and performance in a nuclear criticality safety evaluation process. Whenever possible, the NRC favors use of a quantitative measure to judge implementation of, or compliance with, a principle or methodology. For example, in the case of adherence to the double contingency principle, the SRP requires assignment of specific, quantitative numerical frequencies to each of the controls to determine that a nuclear criticality accident is 'highly unlikely.' The SRP's definition of 'highly unlikely' as a frequency of 10^{-5} is arbitrary and forces differentiation of 10^{-2} and 10^{-3} between two 'unlikely' events in a criticality accident scenario. Measuring compliance to these arbitrary, quantitative values is burdensome and problematic for both licensees and the NRC. In fact, quantification of NRC's expression of the principle of double contingency contradicts guidance of the American National Standard, which upholds the basic definition of the double contingency principle as adequate and sufficient. NEI recommends that industry's current practice of detailed evaluation of credible accident sequences by experienced nuclear criticality engineers continue. Adherence to the ANS-8 guidance should also be continued.

(c) *Graded Level of Protection of Items Relied On For Safety*

The risk of a nuclear criticality for a given accident sequence is established by the ISA. As a nuclear criticality accident is defined to be a high consequence of concern, its risk becomes solely a function of the accident's likelihood of occurrence. Section 70.60(c) requires that graded safety controls (or items relied on for safety) be implemented to prevent nuclear criticality accidents. Systems of controls applied to a high risk sequence should, for example, be inherently more robust than those applied to an intermediate risk sequence. The choice of controls will depend on the risk (or likelihood) of the accident sequence and may include one or more passive engineered, active engineered or administrative controls. While individual controls may vary in their relative importance or robustness, in aggregate, they must render the risk of a nuclear criticality accident to an acceptably (low) level.

§70.60(c) also requires the licensee to ensure that safety controls, or items relied on for safety, to prevent a nuclear criticality accident are *continuously* available and reliable. In practice, specific safety controls will not be operational during maintenance and calibration testing and will not be required when the process is not operational or when Special Nuclear Material (SNM) is not present. The wording of §70.60(c) should, therefore, be modified to address the *risk* of a nuclear criticality accident (rather than its consequences and likelihood) and to assure that items relied on for safety are "…available and reliable *when required to perform their safety functions*."

Section §70.60(c) incorrectly identifies only the likelihood of *external* events as an element of risk from a nuclear criticality accident, thereby excluding the likelihood of process deviations or other internal events as an element of the risk evaluation. In actuality, the likelihood of a process deviation or other *internal* event initiating an accident sequence leading to a potential nuclear criticality is probably far greater than that posed by an external event. The language of §70.60(c) should be clarified.

(d) Nuclear Criticality: Quality Assurance

As noted in comment (a) above, the proposed revisions to Part 70 are likely to be misinterpreted as 'consequence-based' rather than 'risk-based' regulation. An example of this is clearly evident in the draft SRP §5.4.4.1 "Quality Assurance for NCS." Once the appropriate level of protection is afforded to an accident scenario through the identification and application of specific controls, then appropriate assurance must be provided for these controls to ensure their availability and reliability. Draft SRP §5.4.4.1(1) incorrectly requires that all criticality safety controls be afforded the highest level of assurance, while §70.60(d)(3)(vi) and draft SRP §5.4.4.1(5) correctly require the assurance level be commensurate with the importance of the safety function.

Through the application of the double contingency principle, the importance of any single criticality safety control may be less than that of some other control that is the only barrier to an accident with a consequence of concern. Therefore, the highest level of assurance would not necessarily be warranted for criticality controls in accident scenarios with double contingency protection. In addition, the reliability of individual controls should be considered when determining the appropriate level of assurance for criticality safety controls. For example, passive engineered controls in general require a lesser degree of surveillance than active engineered controls or administrative controls. However, each control should be considered relative to the environment and application to determine the level of assurance needed to ensure the reliability and availability of the control.

(e) Historical Nuclear Criticality Data

Part 70 license applications for operating facilities are required by §70.65(c) to include "*…a description of operational events, within the past 10 years, that had a significant impact on the safety of the facility.*" Detailed incident reports of nuclear criticality deviations or violations, including any corrective safety measures that were implemented, are submitted to the NRC at the time of the incident or retained in licensees' records. As the NRC has on file, or available to them, voluminous information on all operational events, including nuclear criticality safety deviations, NEI sees little justification in submitting this information at the time of license application or renewal. NEI recommends that §70.65(c) be deleted from the Part 70 revisions.

II. <u>Concluding Remarks</u>

Nuclear criticality is an important, but readily manageable, operational hazard at fuel cycle facilities. NEI recommends that the proposed revisions of 10 CFR 70 be clarified to reduce their ambiguity and the possibility of interpreting them to be 'consequence-based' rather than 'risk-based' regulations. The rule should permit industry to continue implementation of the double contingency principle as it has done without imposition of a probabilistic methodology. Part 70 should be consistent with American National Standard 8 that upholds the basic definition of the double contingency principle as adequate and sufficient. In support of risk-informed, performance-based regulation, the rule should grant a license applicant the flexibility to implement graded controls (and assurances) based on the results of the ISA.

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