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NUCLEAR ENERGY INSTITUTE

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February 6, 2002

Annette L. Vietti-Cook
Secretary
U.S. Nuclear Regulatory Commission
Mail Stop O-16 C1
Washington, DC 20555-0001

Attention: Rulemakings and Adjudications Staff

Dear Ms. Vietti-Cook:

On behalf of the nuclear energy industry and pursuant to 10 CFR 2.802, NEI submits the enclosed petition to amend 10 CFR 50.46, *Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors*, Appendix A to 10 CFR 50, *General Design Criteria for Nuclear Power Plants*, and Appendix K to Part 10 CFR 50, *ECCS Evaluation Models*. The purpose of the petition is to allow the use of an alternative to the currently required double-ended rupture of the largest pipe in the reactor coolant system in ECCS evaluation models.

The large break loss of coolant accident (LBLOCA) is a central element in the current regulatory framework. It is the basis for numerous regulatory requirements and actions. As such, it dictates the allocation of extensive NRC and licensee resources and attention. Yet, the double-ended rupture is widely viewed as an incredible event of very low safety significance. It diverts attention from more likely, safety-significant postulated events.

The adoption of the proposed amendment will enable technical discussions on redefining the LBLOCA to proceed without being in conflict with the current rules. By amending the regulation in parallel with the technical work, it is estimated that regulatory improvements could be expedited by up to two years.

Template = SECY-051

SECY-02

If you have any questions concerning this petition, please contact me at 202-739-8081 or arp@nei.org.

Sincerely,

Anthony R. Pietrangelo

Anthony R. Pietrangelo

Enclosure

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of a Proposed Rulemaking)
Regarding Amendment of 10 CFR 50.46)
Acceptance Criteria for Emergency Core)
Cooling Systems for Light-Water)
Nuclear Power Reactors)
Appendix A to 10 CFR Part 50)
General Design Criteria for)
Nuclear Power Plants)
Appendix K to 10 CFR 50)
ECCS Evaluation Models) Docket No__________

PETITION FOR RULEMAKING

This petition for rulemaking is submitted pursuant to 10 CFR 2.802 by the Nuclear Energy Institute (NEI) on behalf of the nuclear energy industry. The Petitioner requests that the U.S. Nuclear Regulatory Commission (NRC), following notice and opportunity for comment, amend 10 CFR 50.46, *Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors*, Appendix A to 10 CFR 50, *General Design Criteria for Nuclear Power Plants*, and Appendix K to Part 10 CFR 50, *ECCS Evaluation Models* to allow an alternate break size to the currently required double ended rupture of the largest pipe in the reactor coolant system.

I. STATEMENT OF PETITIONER'S INTEREST

NEI is the organization of the nuclear energy industry responsible for coordinating the combined efforts of all utilities licensed by the NRC to construct or operate nuclear power plants, and of other nuclear industry organizations, in all matters involving generic regulatory policy issues and regulatory aspects of generic operational and technical issues affecting the nuclear power industry. Every entity responsible for constructing or operating a commercial nuclear power plant in the United States is a member of NEI. In addition, NEI's members include major architect / engineering firms and all of the major nuclear steam supply system vendors.

II. BACKGROUND

The specific sections and language of the affected NRC regulations are provided in the Attachment to this petition.

10 CFR 50.46, Appendix A to 10 CFR Part 50, and Appendix K to 10 CFR 50 currently require that a double ended break of the largest pipe in the reactor coolant system be considered in the evaluation of Emergency Core Cooling System (ECCS) acceptance criteria and be used to determine ECCS performance requirements.

In 1967, research results indicated that zircaloy cladding exposed to LOCA like conditions with peak temperatures near 1370°C (well below the zircaloy melting point of 1820°C) could become embrittled and rupture, or even shatter after cooldown. Therefore, a much lower limit on the highest acceptable clad temperature during a LOCA was imposed. In 1971, the AEC issued a policy statement containing interim acceptance criteria for the ECCS for light water reactors. The NRC staff recommended that the ECCS criteria be made more conservative by decreasing the acceptable temperature limit for the cladding from 1260°C to 1204°C (2300°F to 2200°F), and by increasing the conservatism of the methods used to calculate the temperature of the fuel cladding during a LOCA. The revised requirements were published as 10 CFR 50.46, *Acceptance Criteria for Emergency Core Cooling for Light Water Cooled Nuclear Power Reactors*, in 1974.

Appendix K to 10 CFR 50 was promulgated with 10 CFR 50.46 to specify the required and acceptable features of ECCS evaluation models. This appendix was developed with conservative assumptions, and models were required to address areas where data was lacking, or uncertainties were large or unquantifiable at that time.

In 1987, the Commission amended 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 4, *Environmental and dynamic effects design bases, (Fed. Reg. 52 FR 41288)* to exclude from consideration the dynamic effects of postulated ruptures in the Reactor Coolant System (RCS) primary piping and other high energy line piping by the use of methodologies such as leak-before-break (LBB) technology. The justification for the amendment was that the probability of a pipe break in the largest diameter pipe was extremely low for the conditions for which the piping was designed. The GDC 4 amendment allowed the removal of pipe whip restraints and jet impingement devices, and other changes associated with the exclusion of the dynamic effects associated with postulated piping ruptures for piping less than or equal to the largest pipe in the reactor coolant system.

The 1987 GDC amendment introduced an inconsistency into the design basis by allowing the exclusion of the dynamic effects of large postulated pipe ruptures, but retaining the large postulated pipe ruptures for containment design, emergency core cooling, and environmental qualification. In the supplementary information for the 1987 amendment, the Commission acknowledged the inconsistencies and stated that these would be addressed through a long-term evaluation. Now, 14 years after the issuance of the GDC amendment, advances in technology, analytical techniques, and operational feedback have enabled probabilistic fracture mechanics (PFM) methodologies to be further improved. This resulted in NRC approval of a more safety-focused approach for implementing ASME Section XI In-service Inspection requirements, which has significantly improved worker and public safety. These improved methodologies and techniques form the basis for eliminating the inconsistencies introduced in the 1987 amendment. Insights from these new analyses provide the basis for further regulatory improvements through the expanded use of LBB and PFM concepts to the large-break loss of coolant accident (LBLOCA) pipe-break size definition. Such a change, when approved, would focus design and operational procedures and practices on the more likely, safety-significant events. It would ultimately result in additional improvements in the protection of public health and safety and restore consistency to a central element of the regulatory system.

III. PROPOSED ACTION

10 CFR 50.46, Appendix A to 10 CFR Part 50, and Appendix K to 10 Part CFR 50 should be amended to allow the option of considering an alternate maximum break size to the double ended rupture of the largest pipe in the reactor coolant system. The specific regulatory changes are described in subsequent paragraphs.

§50.46 Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.

(c) As used in this section: (1) Loss-of-coolant accidents (LOCA's) are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system, *or up to and including an alternate maximum break size that is approved by the Director of the Office of Nuclear Reactor Regulation.*

Appendix A to Part 50 – General Design Criteria for Nuclear Power Plants

Definitions and Explanations

Loss of coolant accidents. Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system, *or up to and including an alternate maximum break size that is approved by the Director of the Office of Nuclear Reactor Regulation.*

Appendix K to Part 50 – ECCS Evaluation Models

I. Required and Acceptable Features of Evaluation Models.

C. Blowdown Phenomena

1. *Break Characteristics and Flow.* a. In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the primary coolant system, *or up to and including an alternate maximum break area that is approved by the Director of the Office of Nuclear Reactor Regulation.* The analysis shall also include the effects of longitudinal splits in the largest pipes, with the split area equal to the cross-sectional area of the *largest* pipe, *or equal to an alternate maximum break area that is approved by the Director of the Office of Nuclear Reactor Regulation.*

IV RATIONALE FOR THE CHANGE

This petition provides a mechanism for streamlining the regulatory process and improving licensee and NRC focus on matters that have safety significance. US nuclear power plants have attained a very high safety performance record. The insights from probabilistic risk assessments (PRA), from more than 2500 reactor-years of operating experience, and from increased technical knowledge, provide evidence that some systems and design bases events that originally were considered important to safety have significantly less importance than were originally thought, and that some systems or events that were not originally considered important to safety are important.

The LBLOCA is a central element in the design and licensing bases for light water reactors. Advances in analytical techniques (PFM, LBB, and PRA) demonstrate that a LBLOCA, as defined in the regulations, is an extremely unlikely event, which presents negligible risk to public health and safety. The processing and approval of this petition will provide additional confidence to expedite technical discussions on identifying a more probable, safety-significant pipe break size for specific designs. It will enable such discussions to proceed without being encumbered with rulemaking considerations. It will provide added impetus and direction in the development and approval of the LBLOCA implementation applications. As a result, the safety and resource benefits from risk-informing the LBLOCA criterion in §50.46 and Appendices A and K to 10 CFR 50, will be achieved more expeditiously.

Substantial design, licensing, operational activities and resources are expended in addressing this one extremely unlikely event, the instantaneous double-ended break of the largest pipe. As a result, it is appropriate to provide an option for a licensee to revise its design and licensing bases to better focus on the more probable equipment failures and events that have safety significance.

In 1987, the NRC approved the application of LBB methodologies to exclude the dynamic effects of postulated pipe ruptures for various diameters of primary coolant piping, including the main coolant loop. Increased knowledge from advances in technology and analytical techniques coupled with the insights from probabilistic assessments make this petition a natural and correct extension of the currently approved LBB applications. It is noted that in developing the LBB methodology, the LBB acceptance criteria have remained very limiting, and retain conservative margins on leak rate, flaw size, and loads.

This petition is based on a risk informed approach that is derived from the 1995 NRC Policy Statement on Probabilistic Risk Assessment (PRA) *(60 Fed. Reg. 42622, August 1995)*. The policy statement formalized the Commission's commitment to risk-informed regulation through the expanded use of PRA to reduce regulatory conservatisms. The policy statement states, in part, "The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data, and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy."

Following the issuance of the PRA Policy Statement, the NRC developed guidance for making risk-informed, licensee specific changes. Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis* provides metrics on what constitutes an acceptable change. The technical basis for the petition is the insights and information provided in the area of LBB, PFM and licensee specific PRAs. Licensee and generic owners' groups' submittals on associated LBLOCA applications will be based, in part, on these technical insights and information, as well as application specific analyses. As such, the petition and its associated activities are consistent with NRC guidance, policy statements and SECY documents on risk-informing NRC regulations.

In concert with the rulemaking process to implement this petition, the industry will continue to develop the technical work to support the alternate-maximum LOCA break size for varying reactor designs, and start work on the development of specific applications that will be based on the new pipe-break sizes. This technical work will form the basis for industry and regulatory implementation guidelines. These guidelines will provide guidance on the types of changes and the associated methods that can be used, and the level and type of justification necessary to evaluate such changes.

Following licensee implementation of the applications enabled by the approval of this petition, operator and support personnel focus on safety significant matters will be enhanced and plant reliability will be improved. For example, equipment will not be required to meet unnecessarily harsh testing conditions, such as rapid cold starts and loading sequences. Such changes will improve service life and reliabilities, reducing "wear and tear" on safety-significant equipment. Training effectiveness will be improved as operator and plant support staff training and awareness will be focused on the more probable, safety-significant events. Also, operators will no longer have to focus on compliance with Technical Specification limits that are based on margins required for LBLOCA, such as ultimate heat sink temperatures.

This rulemaking will improve the consistency within the existing regulations, and between the regulations and the NRC's reactor oversight process. The petition will provide the basis for an optional, more efficient and integrated approach for resolving current regulatory issues.

The benefits that will be attained through the approval and implementation of the proposals in this petition include: increased plant safety from more realistic Technical Specification surveillance testing and related requirements, such as DG start times, and ultimate heat sink temperature limits; consistency in analytical assumptions; peaking factor increases; and power upratings. Additionally, requirements related to post-LOCA sump boron requirements to maintain core subcriticality with all rods out, and the related potential for sump dilution that could lead to recriticality may be relaxed.

Scope of Rulemaking

This petition retains the LOCA as a design basis event, but redefines the maximum break-size that is subject to a design basis evaluation. The pre-existing LBLOCA analysis will be retained as a historical document, upon adoption of this optional rule change by a licensee. The licensee's plant specific PRA will continue to include LOCAs of all sizes, including a rupture of the largest primary system piping.

It is not the intent of this petition to totally eliminate the mitigation capability associated with a break of the largest pipe in the reactor system. Following approval of a redefined LBLOCA application, a licensee will still retain a capability to mitigate the extremely unlikely break of the largest pipe in the reactor system based on severe accident management principles and activities. Most of the major equipment used to mitigate the existing LBLOCA event is also needed to mitigate other design basis events. As a result, the major components of the current emergency core cooling system (ECCS), such as the high head pumps, intermediate head pumps, and low head pumps, will be retained. However, it should be noted that the system capability and associated requirements and acceptance criteria of these components may be revised, based on the revised maximum LOCA break size, or other design basis accident(s), whichever is more limiting.

The design basis of ECCS components will be based on the revised maximum LOCA break size, or other design basis accident(s), whichever is more limiting. Following approval of the proposed amendments, licensees wishing to apply to use the alternative break-size criteria will amend the applicable safety analyses associated with licensee or owners' group applications. These amended analyses will serve as the basis for the application specific LOCA-related safety analysis assumptions, e.g., control rod insertion following a LOCA and associated post-LOCA sump boron requirements to maintain core subcriticality, containment sump debris generation, and the ultimate heat sink heat removal requirements.

Plants requesting approval of an alternate maximum break size will determine the alternate maximum break size by estimating the appropriate initiating event frequencies for LOCA events and the contribution to overall risk of equivalent break sizes greater than or equal to the alternate maximum break size. Evaluation of the alternate maximum break size will include consideration of defense-in-depth, safety margins, and performance monitoring. For plant changes following NRC acceptance of the use of the alternate maximum break size proposed by the licensee, the risk significance of the changes will be assessed. Such changes will be subject to the change control provisions of 10 CFR 50.59, and may result in a license amendment, if required, in accordance with 10 CFR 50.90.

It is not the intent of this rulemaking petition to be the basis for changing containment structural integrity, such as, changes to design pressure for the containment structure, containment access openings, or containment penetrations. Yet, the petition may result in changes to containment analyses, including the calculation of peak containment accident pressure, subcompartment pressure transients, containment support system requirements, or the environmental qualification temperature profile from a LOCA. Environmental qualification temperature profiles shall continue to consider other design basis breaks in addition to the LOCAs.

V ADDITIONAL CONFORMANCE INFORMATION

Environmental Impact Under NEPA

This petition would not constitute or result in a major Federal action significantly affecting the quality of the human environment. Therefore, an environmental impact statement is not required. The petition will not alter the environmental impact of the licensed activities described in the Final Environmental Impact Statement for each facility, as prescribed in the 1969 National Environmental Protection Act and 10 CFR Part 51. The major components of the ECCS will be retained and licensees will continue to have a capability of mitigating a break of the largest pipe in the reactor coolant system based on severe accident management principles and actions. Through the implementation of the proposals described in this petition, licensees will improve the focus of plant resources and attention on the more probable and safety-significant events.

Paperwork Reduction Act Statement

The proposed petition does not contain any new or amended information requirements that would be subject to the 1980 Paperwork Reduction Act.

Regulatory Backfit Analyses

This petition supports the Commission's goals for improving the efficiency and effectiveness of NRC regulations through risk-informed, performance-based approaches. In response to SECY-98-300, the Commission determined that implementation of risk-informed, performance-based requirements should be optional and voluntary where a regulation already exists. No new requirements would be imposed on a licensee that chooses to implement the petition and, thus a 10 CFR 50.109 backfit evaluation is not applicable.

Regulatory Flexibility Act Certification

The petition will not have an economic impact on a substantial number of small business entities. The companies that own nuclear power plants do not fall within the scope of what constitutes a small business entity as defined in the 1980 Regulatory Flexibility Act, 10 CFR 2.810, and the Small Business Size Standards, 13 Part 121.

VI. CONCLUSION

This petition is consistent with and supports the NRC Strategic and Performance Goals to accomplish the NRC's mission. It is consistent with the Commission's policy statements on PRA and risk-informed, performance-based regulation. Approval and implementation of this petition will improve nuclear safety because a major regulation will be updated to reflect the insights from more than 2500 reactor-years of operation, improvements in engineering knowledge and the insights from probabilistic risk assessments. As a result, plant design, operations and activities, and the associated regulatory oversight will be more focused on events that are more probable and of higher safety significance, while reducing unnecessary regulatory burden.

10 CFR 50.46(a)(1)(i), Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors states:

Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated....(c) As used in this section: (1) Loss-of-coolant accidents (LOCA's) are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.

Appendix A to Part 50 -- General Design Criteria for Nuclear Power Plants states:

(3) Consideration of the type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss-of-coolant accidents. (See Definition of Loss of Coolant Accidents.)

Definitions and Explanations:
Loss of coolant accidents. Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.[(1)]

[1] Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development.

Appendix K to Part 50 -- ECCS Evaluation Models states:

C. Blowdown Phenomena

1. *Break Characteristics and Flow.* a. In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the primary coolant system. The analysis shall also include the effects of longitudinal splits in the largest pipes, with the split area equal to the cross-sectional area of the pipe.