

Fire Response Evaluation

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Agenda

- Introductions
- Opening Remarks
- Refresher on purpose of the report
- Review of NRC Comments 2-8
- Closed Meeting for NRC Comment 1

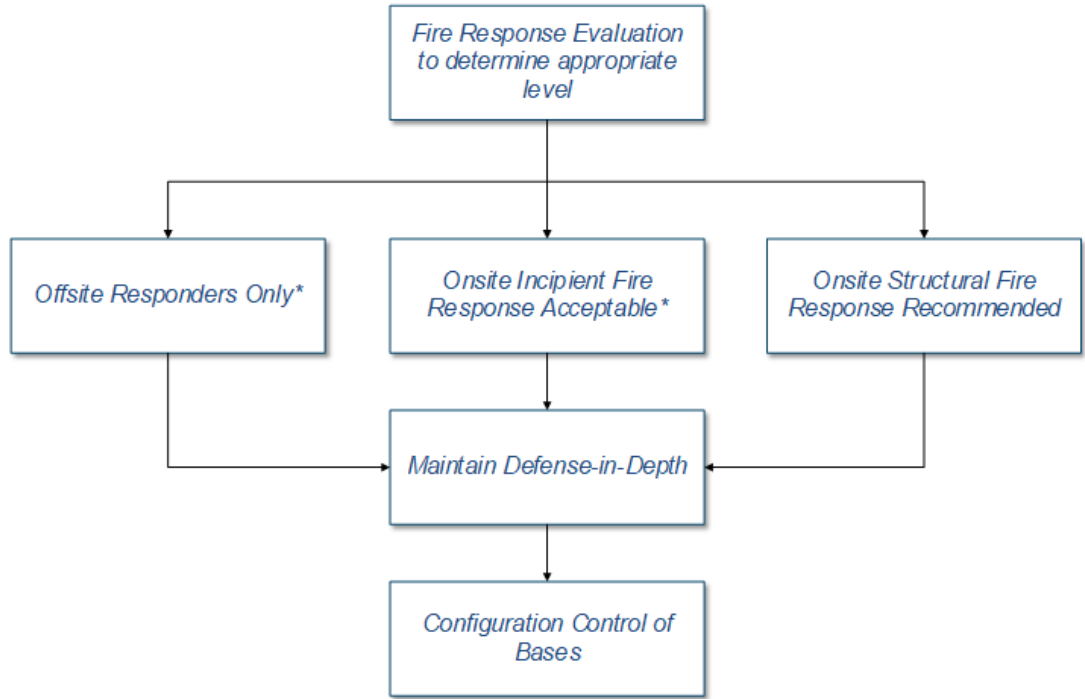
Purpose of the Report

- Provide a process for evaluating the appropriate level of fire response

Offsite Response Only

Onsite Incipient Response

Onsite Structural Response



**Offsite fire response acceptable*

NRC Comments

- Review comments
 - Provide summary of our initial thoughts
 - Any additional NRC thoughts/clarifications

Comments

#	Comment	Path forward
1	<p>General</p> <p>The staff would like to discuss the envisioned analysis steps for a successful implementation of the NEI 24-11 process. For example, is NEI 24-11 intended to get an applicant to a complete fire response methodology? What methodologies do the authors anticipate to address the various evaluation questions? A worked example for each step would be useful.</p>	<p>TerraPower to provide and discuss an example during closed portion of meeting</p>

Comments

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2	<p>General</p> <p>The staff would like to discuss how the NEI 24-11 process would work for a microreactor applicant, with a focus on differences from the process for a larger advanced reactor. For example, a microreactor installed on an exterior concrete pad does not have traditional fire areas with rated barriers and may not have any systems that can be protected in the conventional sense.</p>	<p>A fire protection program, including provisions for manual firefighting, will be established by the applicant at a level of detail commensurate with the scope and risk significance of the facility as described in the overall regulatory submittal. Consistent with 10 CFR 50.48 and GDC/PDC 3, the program will be structured to provide reasonable assurance that the capability to safely shut down the reactor and maintain it in a safe condition is not adversely affected by fires.</p> <p>For microreactors or reactors with comparable risk profiles, the scope, depth, and level of programmatic controls may be appropriately scaled based on the results of the fire hazards and safe shutdown analyses. Where the application demonstrates that fire scenarios do not present credible challenges to nuclear safety functions, fire protection features, systems and programmatic elements - including fire response capabilities - may be reduced or simplified. Such reductions will be justified through engineering analyses and will remain consistent with the treatment of other design and operational attributes that are determined to be of low safety significance for these reactor technologies.</p> <p>Draft 10 CFR 57, Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles, and draft for comment NUREG-2271, Guidelines for Preparing and Reviewing Applications Under 10 CFR Part 57, contain requirements and guidance for microreactor licensing, including information on fire protection. Info to date does not preclude or imply use of offsite fire</p>

Comments

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Comments

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3	<p>General</p> <p>The staff would like to discuss the interplay between the Enhanced Fire Protection Criteria (documented in RG 1.189 Section C.8) and the NEI 24-11 process.</p> <p>Examples:</p> <ul style="list-style-type: none">• It is unclear to the staff what “equipment important to safety” means in the context of the enhanced fire protection criteria.• For a plant that meets the enhanced fire protection criteria, it appears that much of the detailed analysis described in section 5.2, “Fire Hazards and the Potential for Fire Hazards to Rapidly Grow and Damage Systems Important to Safety,” may be unnecessary because no SSCs needed to safely shut the plant down could be targets as described in this section because they would be located in a separate fire area.	<p>Given different reactor types, we would expect a compliance/alignment document to be submitted as part of the operating license, which can follow different formats.</p> <p>A fire protection program that meets PDC-3/10 CFR 50.48(a) will need to be demonstrated as part of the operating license submittal, and the manual fire fighting response will be one of the elements of the fire protection program. It is not intended or expected that the manual fire fighting response discussion be a dominant area of discussion. Instead, it would be part of an integrated discussion on the overall acceptability of the fire protection program.</p> <p>We acknowledge that a range of terminology has historically been used to describe nuclear safety objectives associated with fire protection (e.g., “safe shutdown,” “post-fire safe shutdown capability,” or “important-to-safety functions”). For advanced reactors, applicants are expected to define a technology-specific “safe state” consistent with the plant design and licensing basis. As such, terminology used within the fire protection program should be clearly defined and consistently applied.</p>

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Comments

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4	<p>Section 6.2.3</p> <p>The staff would like to discuss the applicability of data collected from the current fleet to designs where the basic conditions differ (for example, staffing levels may be significantly lower at an advanced reactor, and this influences the timing of when a particular fire would be detected and likelihood that an incipient fire brigade would be able to extinguish it). How can the conclusion in section 6.2.3, under the question “Is the fire expected to be identified in the incipient stage?” be justified using data that does not appear to reflect the conditions at the plant?</p>	<p>Like many aspects of advanced reactor design and operation, our most available data is from the LWR operating fleet. Data related to detection and extinguishment of fires is no different. The use of current fleet fire event data in NEI 24-11 Rev. 1 is appropriate because:</p> <ul style="list-style-type: none">• It is not used in a direct correlation advanced reactor conditions, using best available information as an “example”• The determination of incipient stage fire detection is based primarily on plant-specific design and analysis• The methodology of NEI 24-11 explicitly accounts for reduced staffing and different operational concepts• Conservative assumptions are maintained where uncertainties exist.

Comments

#	Comment	Path forward
5	<p>Section 3.1.3 Regulatory Guide (RG) 1.189, “Fire Protection for Nuclear Power Plants”, is listed in section 3.1 (Fire Protection Program Regulatory Requirements). RG 1.189 is not a regulatory requirement and should be listed in Section 3.2 (NRC Fire Protection Guidance for Power Reactors).</p>	<p>I think this comment intended to be RG 1.232. Moved to Section 3.1.3 to Section 3.2.2 and provided a link from 3.12. See comment 6.</p>
6	<p>Section 3.1 RG 1.232, “Principal Design Criteria for Non-Light-Water Reactors”, is listed in section 3.1 (Fire Protection Program Regulatory Requirements). RG 1.232 is not a regulatory requirement and should be listed in Section 3.2 (NRC Fire Protection Guidance for Power Reactors). There does not appear to be a way to get to a Structural Fire Brigade result except through some insufficiency in offsite response.</p>	<p>Deleted reference to RG 1.232 in Section 3.1 bulleted list</p>

Comments

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7	Section 3.3 Consider adding a note that indicates ANSI/ANS 15.17, “Fire Protection Program Criteria for Research Reactors” has been withdrawn and may contain outdated information.	Added sentence to bulleted list in Section 3.3. Note the latest version of NUREG-1537 Part 1 references the ANSI/ANS standard.

Comments

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8	<p>Section 6.2.1</p> <p>As a comment on how the eventual endorsement may be qualified:</p> <p>This section contains the statement: “Given that meeting this goal [the enhanced fire protection criteria for advanced reactors] is a prerequisite for the design, radiological release is not postulated due to fire damage.”</p> <p>Because the enhanced criteria requires that there is a success path somewhere outside the fire area that is free of fire damage, therefore assuring safe shutdown, the staff does not object to this statement with respect to deterministic space for the purposes of the fire response evaluation, provided that a thorough post-fire safe shutdown analysis is successfully performed, including consideration of associated circuits, multiple spurious operations, and other circuit analysis.</p> <p>However, this is not the case in probabilistic space, and every plant is expected to have a non-zero Core Damage Frequency (CDF) and Large Early Release Frequency (LERF), or design-specific equivalent metrics.</p>	<p>The statement in NEI 24-11 is not intended to imply zero fire risk. It does not preclude non-zero contributions to:</p> <ul style="list-style-type: none"> • Core Damage Frequency (CDF) • Large Early Release Frequency (LERF), or • Equivalent advanced reactor metrics <p>This distinction is consistent with existing risk-informed programs. Deterministic analyses define design-basis capability. PRA evaluates residual risk beyond design-basis assumptions</p> <p>The NEI 24-11 approach is particularly appropriate because advanced reactors often:</p> <ul style="list-style-type: none"> • Rely on passive safety functions • Have limited safety-related SSCs • Exhibit low combustible inventories

Comments

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Next Steps

- Publish NEI 24-11 Rev. 2 and submit to NRC for endorsement
- Interim approval or confidence measure before Reg. Guide revision?