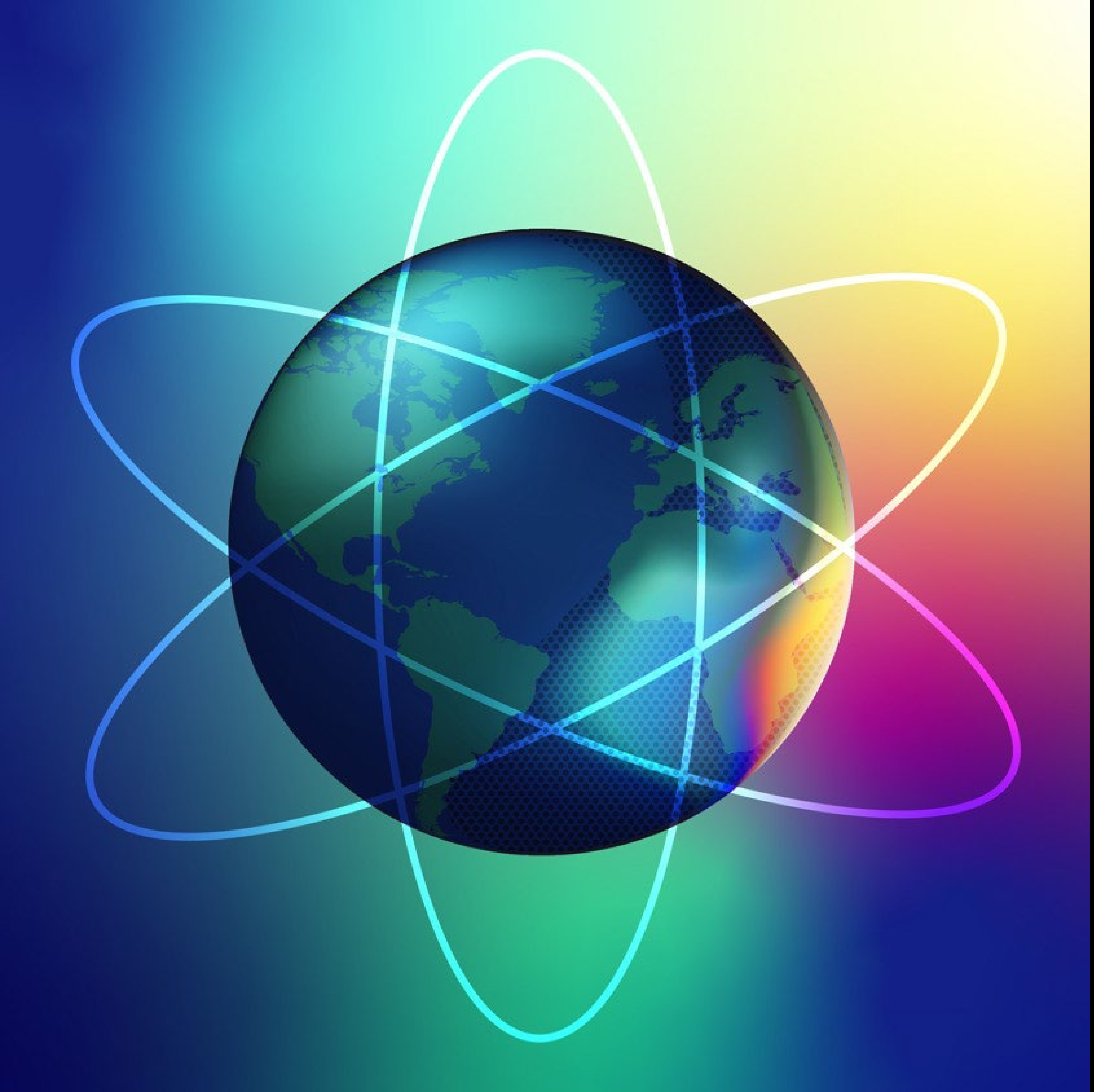

NEI NUCLEAR ENERGY INSTITUTE

ANS Community of Practice: Risk-Informing Codes and Standards

August 30, 2024

Topics

1) Differences between LMP/TICAP Terminology and Traditional Language Used in Codes

2) High Level Examples of Issues

3) IEEE 497 as a Case Study for Risk-Informing a Standard

TICAP / ARCAP vs 10 CFR 50.2: SR Definition

10 CFR 50.2

- Safety-related structures, systems and components means those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:
- (1) The integrity of the reactor coolant pressure boundary
- (2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

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- SSCs selected by the designer from the SSCs that are available to perform the RSFs to mitigate the consequences of DBEs to within the LBE F-C Target, and to mitigate DBAs that only rely on the SR SSCs to meet the dose limits of 10 CFR 50.34 using conservative assumptions
- SSCs selected by the designer and relied on to perform RSFs to prevent the frequency of BDBE with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C Target

Takeaways

- Can justify NSR reactor coolant pressure boundary and reactor building (potentially RPS too). Implications for C&S and other STs needs work.

TICAP / ARCAP vs SRP: LBE Selection

Table 3-1. Definitions of Licensing Basis Events

Event Type	Current Definition or Common Use	Guidance Document Definition
Anticipated Operational Occurrences (AOOs)	*"Conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power."** [SRP 15.0 and 10 CFR 50 Appendix A]	Anticipated event sequences expected to occur one or more times during the life of a nuclear power plant, which may include one or more reactor modules. Event sequences with mean frequencies of 1×10^{-2}/plant-year and greater are classified as AOOs. AOOs take into account the expected response of all SSCs within the plant, regardless of safety classification.
Design Basis Events (DBEs)	*"Conditions of normal operation, including AOOs, design-basis accidents, external events, and natural phenomena, for which the plant must be designed to ensure functions of safety-related electric equipment that ensures the integrity of the reactor coolant pressure boundary; the capability to shut down the reactor and maintain it in a safe shutdown condition; or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures."* [SRP 15.0]	Infrequent event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than AOOs. Event sequences with mean frequencies of 1×10^{-4}/plant-year to 1×10^{-2}/plant-year are classified as DBEs. DBEs take into account the expected response of all SSCs within the plant regardless of safety classification.
Beyond Design Basis Events (BDBEs)	*"This term is used as a technical way to discuss accident sequences that are possible but were not fully considered in the design process because they were judged to be too unlikely. (In that sense, they are considered beyond the scope of design-basis accidents that a nuclear facility must be designed and built to withstand.) As the regulatory process strives to be as thorough as possible, 'beyond design-basis' accident sequences are analyzed to fully understand the capability of a design."* [NRC Glossary]	Rare event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than a DBE. Event sequences with mean frequencies of 5×10^{-7}/plant-year to 1×10^{-4}/plant-year are classified as BDBEs. BDBEs take into account the expected response of all SSCs within the plant regardless of safety classification.
Design Basis Accidents (DBAs)	*"Postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components."* [SRP 15.0] *"A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety."* [NRC Glossary and NUREG-2122]	Postulated event sequences that are used to set design criteria and performance objectives for the design of Safety Related SSCs. DBAs are derived from DBEs based on the capabilities and reliabilities of Safety-Related SSCs needed to mitigate and prevent event sequences, respectively. DBAs are derived from the DBEs by prescriptively assuming that only Safety Related SSCs are available to mitigate postulated event sequence consequences to within the 10 CFR 50.34 dose limits.
Licensing Basis Events (LBEs)	Term not used formally in NRC documents.	The entire collection of event sequences considered in the design and licensing basis of the plant, which may include one or more reactor modules. LBEs include AOOs, DBEs, BDBEs, and DBAs.

- Provides a Technology-Inclusive Method for establishing LBEs
- Provides a method for not having to analyze low frequency events (no LLOCA)
- Leads to some challenges in Regulations and C&S that rely on the traditional definitions

TICAP / ARCAP vs SRP: Safety Function

<u>10 CFR 50</u>

- Safety Function is not defined in 10 CFR 50.2, but is used extensively in 10 CFR 50 (Appendix A, 10 CFR 50.34, 50.49)

<u>NRC Glossary</u>

- Examples of safety-related functions include shutting down a nuclear reactor and maintaining it in a safe-shutdown condition.

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LMP Term	Acronym	Definition	Source
Terms Associated with Functions			
Fundamental Safety Function	FSF	Safety functions common to all reactor technologies and designs; includes control heat generation, control heat removal and confinement of radioactive material	IAEA-TECDOC-1570
PRA Safety Function	PSF	Reactor design specific SSC functions modeled in a PRA that serve to prevent and/or mitigate a release of radioactive material or to protect one or more barriers to release. In ASME/ANS-Ra-S-1.4-2013 these are referred to as "safety functions." The modifier PRA is used in the LMP GD to avoid confusion with safety functions performed by Safety-Related SSCs.	LMP, ASME/ANS-Ra-S-1.4-2013
Required Functional Design Criteria	RFDC	Reactor design-specific functional criteria that are necessary and sufficient to meet the RSFs	LMP

Other Issues introduced by LMP / 50.69

10 CFR 50.69 (b)

- Allows alternate treatments to:
  - 10 CFR 21
  - 10 CFR 50.49
  - 10 CFR 50.65
  - Appendix B
  - 10 CFR 50.46a(b)
  - 10 CFR 50.55(e)
  - 10 CFR 50.55a(f)
  - 10 CFR 50.55a(g)
  - 10 CFR 50.72
  - 10 CFR 50.73
  - Parts of Appendix J
  - Some Seismic Requirements

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- From RG 1.233 endorsing NEI 18-04: "The staff finds that the NEI 18-04 methodology, including assessments of event sequences and DID, obviates the need to use the single-failure criterion as it is applied to the deterministic evaluations of AOOs and DBAs for LWRs."
- Table 4-1 provides Special Treatment Guidance that could be used a technical basis for alterative treatments for Appendix B, 50.65, 50.49 and more.

Examples of Code Intersections with LMP

- Stress analysis per ASME Section III Division 5 relies of the definitions of Anticipated Operating Occurrences, Design Basis Events and that requires interpretation for those following LMP.
- IEEE603, IEEE497 and others emphasize Single Failure Criteria (SFC) and require significant edits to address flexibility.
- Many codes are prescriptive in their special treatments generally. ASME NQA-1 addressed flexibility from 50.69 in new SubPart 3.1-2.5, IEEE developed IEEE 1819 to address 50.69 and is using it as a reference to inform other C&S.
    - Much more work to be done for LMP, will need to be driven by first mover's
- ASME Section XI Div 2, ANS 2.26 and ASCE 43-19 informed by LMP
- LMP and TRISO can lead to NSR pressure boundaries and reactor buildings that are not containment.

Case Study: IEEE497

- Definitions are difficult and important. IEEE coordinated with IEC and tweaked definitions and added terms (Design extension conditions or DECs) to ensure the Code definitions aligned with all potential uses (traditional US, LMP, International)
  - Example: The definition of "safety system" was tweaked from the 50.2 definition to a more technology-inclusive definition
- Codes should be searched for areas of prescription and changed to allow flexibility.
  - Example: Type B variables were defined based on performing the traditional 50.2 SR functions, draft changes these to safety functions defined in the licensing basis.
  - Example: Type C variables were defined as those monitoring fission product barriers (fuel cladding, reactor coolant pressure boundary and containment pressure boundary). Draft changes the language to "safety-related fission product barriers" leaving the past barriers as an example.

Case Study: IEEE497

- Section 6 of IEEE497 covers design criteria and has prescriptive requirements not in line with LMP or 50.69 (SFC, separation, SR power supply, equipment qualification, seismic design etc).
  - Example: Draft IEEE497 addresses this with language in the front-matter: “Facilities adopting a risk-informed program for equipment categorization and treatment (in accordance with regulatory approval) shall implement this standard in conjunction with risk informed methods such as that defined in IEEE Std 1819™.”
  - The above approach allows traditional plants to follow the guidance without change, but for those who choose a risk-informed approach they can pick and choose the guidance as justified by their design and approved in their licensing basis.

Conclusions

- There is significant work to do to risk-inform standards
- SDOs can no longer rely on traditional definitions of “Safety-Related”, “Safety Function”, “AOOs”, “DBEs”, “DBAs”, etc.
- Language should be inclusive of licensing approaches: Traditional, 50.69, LMP, etc.
- Work has started and some examples are out there: IEEE497 in draft, ASME Section XI Div 2, ANS 2.26, ASCE 43-19, OM-2, SubPart 3.1-2.5 of NQA-1
- Industry needs to be very careful of scope creep; NSRST and RISC-2 SSCs should not receive SR ST
- Code updates will require a whole industry effort!

Conclusions

- NEI, SDOs, EPRI, IAEA, ANS, DOE are involved in an Advanced Reactor Codes and Standards Collaborative (ARCSC) trying to drive this issue.
- Tuesday ARCSC Workshop 9/10 at Nuclear Energy Assembly
- Mark Richter (QA), Tom Basso(ASME), Alan Campbell (I&C) and Jon Facemire (LMP) at NEI are available for support and collaboration
- The ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM) Sub-Committee on Risk Applications (SCORA) is available to support Risk-Informing Standards
- ANS RP3C has put out a guidance document for "Incorporating Risk-Informed and Performance-Based Approaches/Attributes in ANS Standards"
- NRC Standards Forum September 25
- The real work happens on Standard Committees, join, engage, collaborate and let's develop a comprehensive set of risk-informed standards for industry!