Advanced Reactor Stakeholder Public Meeting

May 27, 2021

Microsoft Teams Meeting
Bridgeline: 301-576-2978
Conference ID: 550 337 464#

Time	Agenda	Speaker
10:00 – 10:10 am	Opening Remarks	NRR/DANU
10:15 - 10:30 am	ASME Section III, Division 5 Design Tool Software	RES
10:30 - 10:45 am	Revision to NRC Pre-application Engagement White Paper	NRR/DANU
11:00 - 11:15 am	Inspection and Oversight Framework for Advanced Reactors	NRR/DANU
11:15 - 11:30 am	Export Controls Report on How Advanced Reactors fit within Part 110	OIP
11:30 am – 1:00 pm	BREAK	All
1:00 - 2:30 pm	Graded Probabilistic Risk Assessment (PRA) Approach for Advanced Reactors	NRR/DANU
2:30 - 3:15 pm	White Paper on draft Licensing Modernization Project (LMP)-based Technical Specification Guidance	NRR/DANU
3:15 – 3:30 pm	Concluding Remarks and Future Meeting Planning	NRC/All

Advanced Reactor Integrated Schedule of Activities

Advanced Reactor - Summary of Integrated Schedule and Regulatory Activities

Summary of Integrated Schedule and Regulatory Activities (updated 11/02/2020)

Advanced Reactor Program - Summary of Integrated Schedule and Regulatory Activities*

Strategy 1	Knowledge, Skills, and Capability
Strategy 2	Computer Codes and Review Tools
Strategy 3	Flexible Review Processes
Strategy 4	Consensus Codes and Standards
Strategy 5	Policy and Key Technical Issues
Strategy 6	Communication

Legend

- Concurrence (Division/Interoffice)
- Federal Register Publication
- Public Comment Period
- Draft Issuance of Deliverable
- Final Issuance of Deliverable
- • EDO Concurrence Period
- Commission Review Period**
- ▼ ACRS SC/FC (Scheduled or Planned)
- External Stakeholder Interactions
- ↓ Public Meeting (Scheduled or Planned)

Version 11/2/20

Present Day

Strategy	Regulatory Activity	Commission Papers	Guidance	Rulemaking	NEIMA	Complete	2020												2021											
							Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
1	Development of non-Light Water Reactor (LWR) Training for Advanced Reactors (Adv. Rxs) (NEIMA Section 103(a)(5))																													
	FAST Reactor Technology				x	x																								
	High Temperature Gas-cooled Reactor (HTGR) Technology				x	x																								
	Molten Salt Reactor (MSR) Technology				x	x																								
	Competency Modeling to ensure adequate workforce skillset					x																								
	Identification and Assessment of Available Codes					x																								
	Development of Non-LWR Computer Models and Analytical Tools																													
	Code Assessment Reports Volumes 1 (Systems Analysis)					x																								
	Reference plant model for Heat Pipe-Cooled Micro Reactor					x																								
	Reference plant model for Sodium-Cooled Fast Reactor					x																								
	Reference plant model for Fluoride-Salt-Cooled High-Temperature Reactor																													
	Reference plant model for Gas-Cooled Pebble Bed Reactor																													
	Reference plant model for Molten Salt Fueled Reactor																													
	Code Assessment Reports Volumes 2 (Fuel Perf. Anaylsis)					x																								
	FAST code assessment for metallic fuel																													
	FAST code assessment for TRISO fuel																													
	Code Assessment Reports Volumes 3 (Source Term Analysis)					x																								

https://www.nrc.gov/reactors/new-reactors/advanced/details#advSumISRA

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ASME Section III, Division 5 Design Tool Software

Advanced Reactors Stakeholders Meeting

May 27, 2021

Jeff Poehler
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RES/DE/REB
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Background

- Verifies construction rules for high-temperature components used in ANLWR designs.
- Enables staff to perform confirmatory analysis of ANLWR component designs.
- Software is publicly available and could be used by ANLWR designers.
- Developed under contract by Argonne National Laboratory.

What the tool does

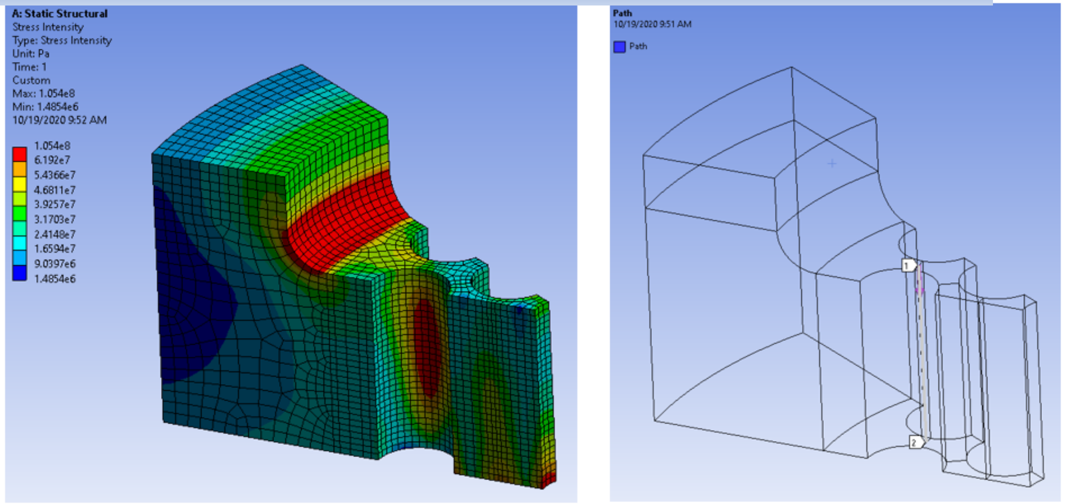
- **The software executes the Section III, Division 5 design checks for:**

Primary Load Limits	Strain limits	Creep-Fatigue	Elastic-Perfectly Plastic
• HBB-3000 • Load Controlled Rules • Design Loading • Service Level loadings • Service Life Fraction	• Deformation controlled rules • Using elastic analysis (HBB-T-1320) and simplified inelastic analysis (HBB-T-1330)	• Using Elastic Analysis (HBB-T-1430) • Does not perform inelastic analysis (HBB-T-1420)	• Code Case N-861 strain limits • Code Case N-862 Creep-Fatigue

Using the Software

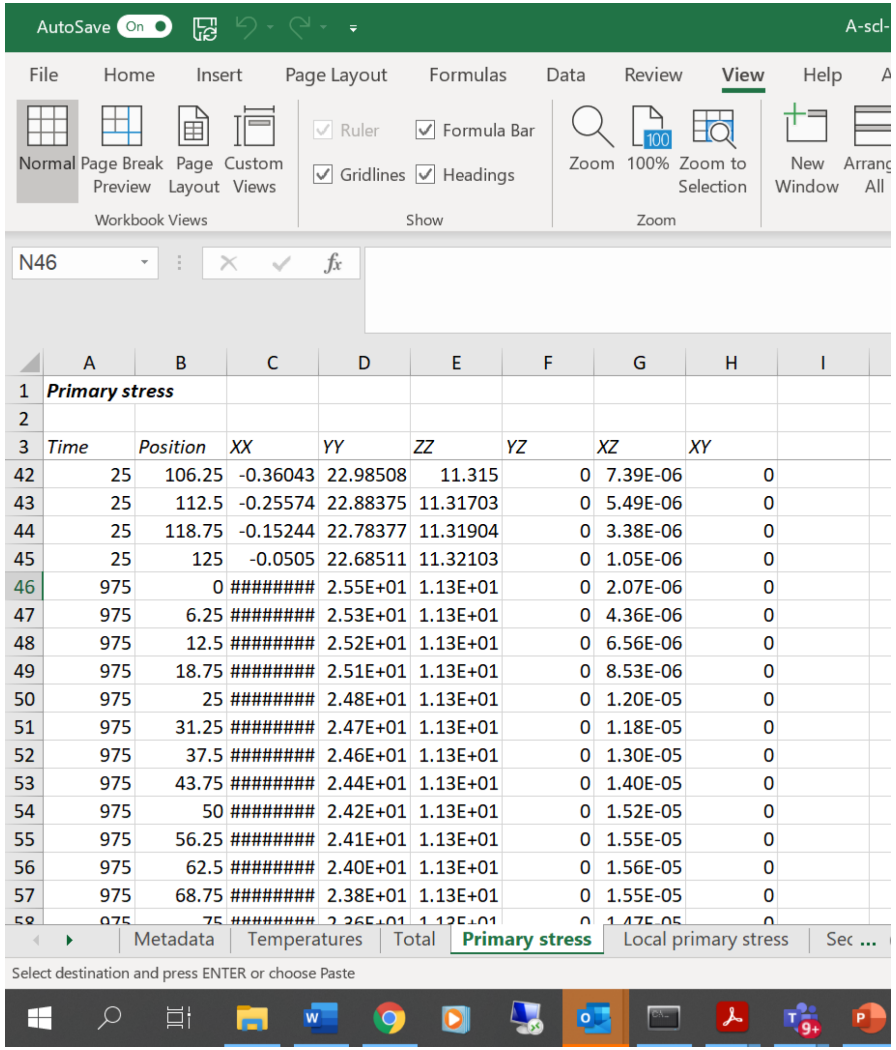
- Users must have Python 3.7 or higher installed on their machine. Available from Anaconda/Python https://www.anaconda.com/products/individual
- Software consists of two modules:
  - hbbdata – contains the allowable stresses and properties for the five Section III, Division 5 Class A materials plus Alloy 617
  - hbbdata – executes the design checks
- Stress inputs may be from any commercial finite element analysis software
  - Must be entered in a spreadsheet in standard format.

1. Design – finite element analysis results

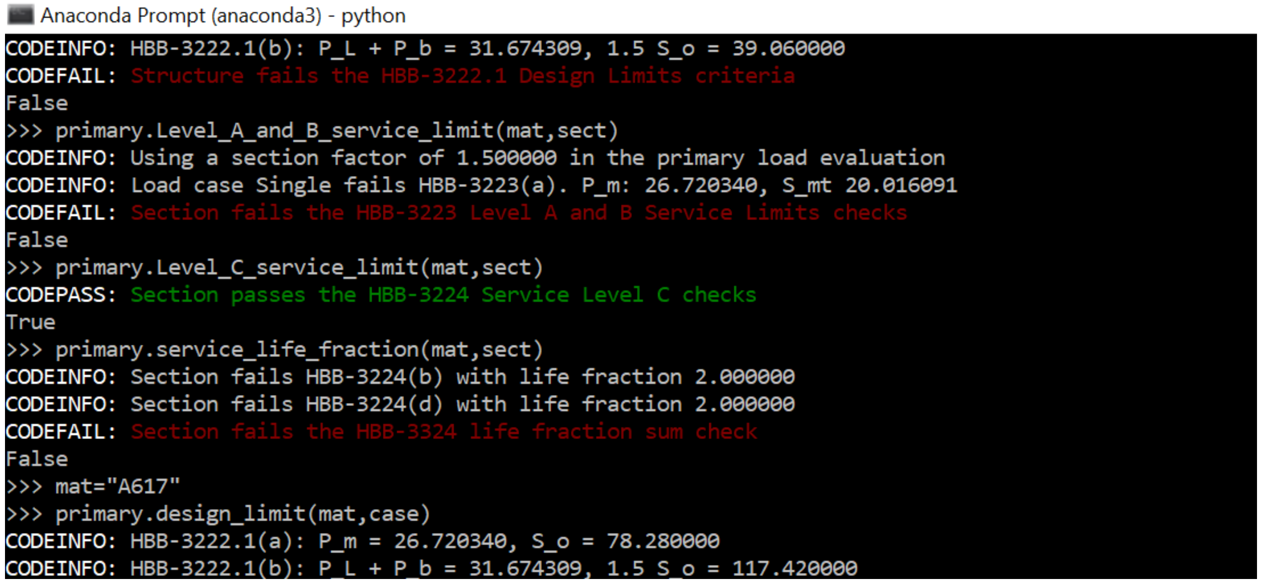


(a) (b)

2. FEA results in Excel spreadsheet format



	A	B	C	D	E	F	G	H
1	*Primary stress*							
2								
3	*Time*	*Position*	*XX*	*YY*	*ZZ*	*YZ*	*XZ*	*XY*
42	25	106.25	-0.36043	22.98508	11.315	0	7.39E-06	0
43	25	112.5	-0.25574	22.88375	11.31703	0	5.49E-06	0
44	25	118.75	-0.15244	22.78377	11.31904	0	3.38E-06	0
45	25	125	-0.0505	22.68511	11.32103	0	1.05E-06	0
46	975	0	########	2.55E+01	1.13E+01	0	2.07E-06	0
47	975	6.25	########	2.53E+01	1.13E+01	0	4.36E-06	0
48	975	12.5	########	2.52E+01	1.13E+01	0	6.56E-06	0
49	975	18.75	########	2.51E+01	1.13E+01	0	8.53E-06	0
50	975	25	########	2.48E+01	1.13E+01	0	1.20E-05	0
51	975	31.25	########	2.47E+01	1.13E+01	0	1.18E-05	0
52	975	37.5	########	2.46E+01	1.13E+01	0	1.30E-05	0
53	975	43.75	########	2.44E+01	1.13E+01	0	1.40E-05	0
54	975	50	########	2.42E+01	1.13E+01	0	1.52E-05	0
55	975	56.25	########	2.41E+01	1.13E+01	0	1.55E-05	0
56	975	62.5	########	2.40E+01	1.13E+01	0	1.56E-05	0
57	975	68.75	########	2.38E+01	1.13E+01	0	1.55E-05	0



Anaconda Prompt (anaconda3) - python

```
CODEINFO: HBB-3222.1(b): P_L + P_b = 31.674309, 1.5 S_o = 39.060000
CODEFAIL: Structure fails the HBB-3222.1 Design Limits criteria
False
>>> primary.Level_A_and_B_service_limit(mat,sect)
CODEINFO: Using a section factor of 1.500000 in the primary load evaluation
CODEINFO: Load case Single fails HBB-3223(a). P_m: 26.720340, S_mt 20.016091
CODEFAIL: Section fails the HBB-3223 Level A and B Service Limits checks
False
>>> primary.Level_C_service_limit(mat,sect)
CODEPASS: Section passes the HBB-3224 Service Level C checks
True
>>> primary.service_life_fraction(mat,sect)
CODEINFO: Section fails HBB-3224(b) with life fraction 2.000000
CODEINFO: Section fails HBB-3224(d) with life fraction 2.000000
CODEFAIL: Section fails the HBB-3324 life fraction sum check
False
>>> mat="A617"
>>> primary.design_limit(mat,case)
CODEINFO: HBB-3222.1(a): P_m = 26.720340, S_o = 78.280000
CODEINFO: HBB-3222.1(b): P_L + P_b = 31.674309, 1.5 S_o = 117.420000
```

3. Python – import results and run checks

What the Tool Includes

hbbanalysis	hbbdata
• Python scripts • Excel templates for FEA results input • Excel files with example FEA results • Documentation	• Python scripts • Text files with materials data (hbbdata) • Documentation

Obtaining the Software

- Fill out non-disclosure agreement (NDA) Form
- Mail or email to safetycodes@nrc.gov.
- NRC staff will review and determine if the software can be distributed to the requester.
- If approved, NRC staff will send a link to download the package from Box.
- More information at https://www.nrc.gov/about-nrc/regulatory/research/obtainingcodes.html, or contact:
    - jeffrey.Poehler@nrc.gov

References

- User manuals are available at ML21050A044 (ADAMS Package)
  - HBBdata Documentation, Release 1.0, Argonne National Laboratory (ML21050A042)
  - HBBanalysis Documentation, Release 1.0, Argonne National Laboratory (ML21050A041)
- NRC Public Web Site information on the ASME Section III, Division 5 Design Tool – https://www.nrc.gov/about-nrc/regulatory/research/safetycodes.html
- NDA Form - https://www.nrc.gov/docs/ML1523/ML15233A353.pdf https://www.nrc.gov/docs/ML1523/
- Obtaining Python - https://www.anaconda.com/products/individual

Draft White Paper - Preapplication Engagement to Optimize Advanced Reactors Application Reviews

Benjamin Beasley, Branch Chief
Advanced Reactor Licensing Branch

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Pre-Application Engagement

- NRC staff applied a graded approach to identify key safety and environmental licensing areas for pre-application engagement with advanced reactor developers
  - Topical Reports - definitive findings
  - White Papers, Audits and Meetings – feedback and staff awareness
- Program is voluntary

Benefits of Pre-Application Engagement

- Enhanced regulatory predictability
- Greater review efficiency
- More visibility for public on key topics
- Early engagement and interactions with ACRS and other agencies

Benefits of Full Execution of White Paper Pre-Application Engagement

- Review schedule at least 6 months shorter than the generic schedules depending on the complexity of the design
- Acceptance review completed in two weeks, only addressing administrative aspects (e.g., proprietary review, making the application publicly available, and issuing notice of availability)
- Key Assumptions for shortened schedule
  - Timely Responses to Requests for Additional Information (RAIs)
  - No Substantive Changes to Application (unless driven by RAIs)
  - No Significant Design Changes (Pre-application vs Application)

Summary of Key Changes

- Previous (January) version - ADAMS Accession No. ML21014A267
- New version –ADAMS Accession No. ML21145A106
- Topical Reports Section
  - Added discussion on how topical reports would still benefit construction permit applicants
- Fuel qualification and testing
  - Aligned information with NRC's Fuel Qualification for Advanced Reactors draft white paper (ADAMS Accession No. ML20191A259)
- Safety and accident analyses methodologies and associated validation
  - Specified that the test program for verification and validation of the engineering computer programs should satisfy the requirements in 10 CFR 50.43(e)

Summary of Key Changes

- Regulatory Gap Analysis Report
  - Should list Part 50 or 52 requirements for which an exemption, case-specific order, or rule of particular applicability would be sought
    - Consistent with NRC's draft white paper, "Analysis of Applicability of NRC Regulations for Non-Light Water Reactors"
- Identification and justification of the use of engineering computer programs
  - After further consideration, this was not deemed necessary for this voluntary pre-application program and was therefore, deleted.

Questions and Comments

Advanced Reactor Construction Inspection and Oversight (ARCOP) Framework

May 27, 2021

Framework

- Establish scope
  - Construction Oversight and Operational Oversight
  - Advanced Reactors
- Vision and strategy
- Expectations and considerations
- Identify attributes of program

Note: specific procedures and performance indicators will be developed in later phases of ARCOP effort

Advanced Reactor Construction Inspection and Oversight Framework

- Advanced Reactors definition includes non-light water reactors (non-LWRs), small modular reactors (SMRs), and fusion
- Wide range of non-LWR technologies being pursued by vendors (e.g., liquid sodium cooled, high-temperature gas cooled, heat pipe, etc.)
- License applications are likely to be risk-informed, performance based (e.g., using RG 1.233 endorsed or similar process)
- DOE’s Advanced Reactor Demonstration Project (ARDP) awards has provided a level of commitment and schedule certainty for additional near-term applications
- Prudent to begin work on developing ARCOP framework

ARCOP Framework Development Considerations:

- Existing Reactor Oversight Process (ROP) is based on LWRs but is risk-informed and could be leveraged for ARCOP
  - https://www.nrc.gov/reactors/operating/oversight.html
- Existing Construction Reactor Oversight Process (cROP) is also based on LWRs and was specifically developed to support new reactors licensed under the Part 52 process
  - https://www.nrc.gov/reactors/new-reactors/oversight/crop.html
- Similar to the effort for new reactors, a new framework needs to be developed to support Advanced Reactors
- NRC effort to develop an outline for an ARCOP framework was recently initiated

Broad Landscape of Advanced Reactor Designs

Liquid Metal Cooled Fast Reactors (LMFR)

Sodium-Cooled
- TerraPower/GEH (Natrium)*
- GEH PRISM (VTR)
- Advanced Reactor Concepts

Lead-Cooled
- Westinghouse
- Columbia Basin
- Hydromine

High-Temperature Gas-Cooled Reactors (HTGR)

TRISO Fuel
- X-energy *
- Framatome
- StarCore
- MIT

Other:
- General Atomics (EM2)
- General Atomics

Molten Salt Reactors (MSR)

Liquid Salt Cooled
- Kairos *
- Kairos (Hermes|RTR)

Liquid Salt Fueled
- Terrestrial *
- TerraPower
- Southern (TP MCFR) |RTR
- ACU |RTR *
- Elysium
- Thorcon
- Muons
- Flibe
- Alpha Tech

Micro Reactors

Transportable
- Westinghouse (eVinci)
- BWX Technologies
- X-energy
- Radiant |RTR

Stationary
- Ultra Safe |RTR
- Oklo

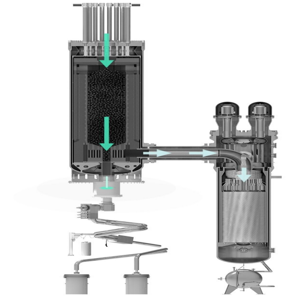
LEGEND

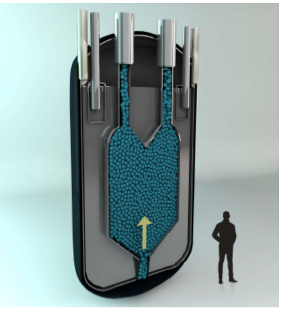
ARDP Awardees
- Demo Reactors
- Risk Reduction
- ARC-20

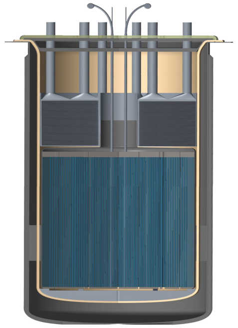
In Licensing Review

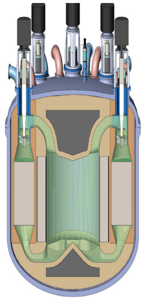
* Preapplication

RTR Research/Test Reactor









OKLO

X energy

Kairos Power

TERRESTRIAL ENERGY

TerraPower
A Nuclear Innovation Company

Active NRC Regulatory Engagements on Advanced Reactors (non-LWR designs)

- OKLO – custom Combined License application review
- Current pre-application interactions
  - X-Energy (ARDP awardee)
  - TerraPower (ARDP awardee)
  - Kairos Power – topical reports
  - Terrestrial Energy USA
- International cooperation with CNSC

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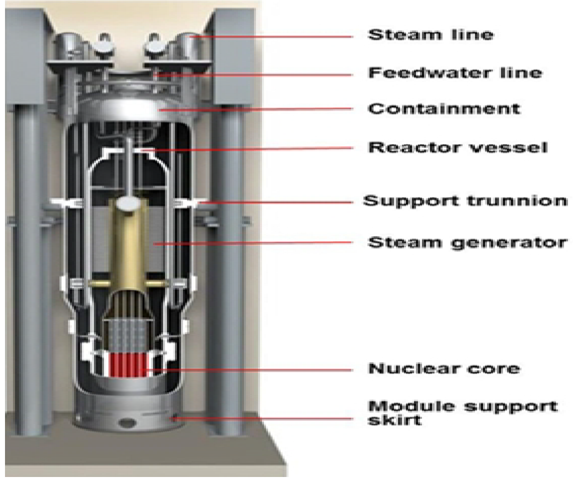
CNSC CCSN

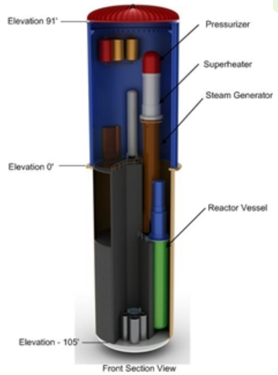
Active NRC Regulatory Engagements on LWR-SMRs

- TVA – Clinch River Early Site Permit review completed
- NuScale – recently completed Design Certification review
- Pre-application interactions
    - NuScale SDA – topical reports
    - GEH BWRX-300 – topical reports
    - Holtec SMR-160 – topical reports

HITACHI



NUSCALE



HOLTEC INTERNATIONAL

TVA Tennessee Valley Authority

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Licensing Modernization Project:

A risk-informed, consequence-oriented approach to establish licensing basis and content of applications

(see Regulatory Guide 1.233 https://www.nrc.gov/docs/ML2009/ML20091L698.pdf)

Focus of LMP:

- Risk-informed selection of Licensing Basis Events (LBEs)
- Determination of safety classification of SSCs
- Defense in depth adequacy assessment (i.e., plant capability, programmatic, risk-informed)
- Determination of special treatments for non-safety-related SSCs

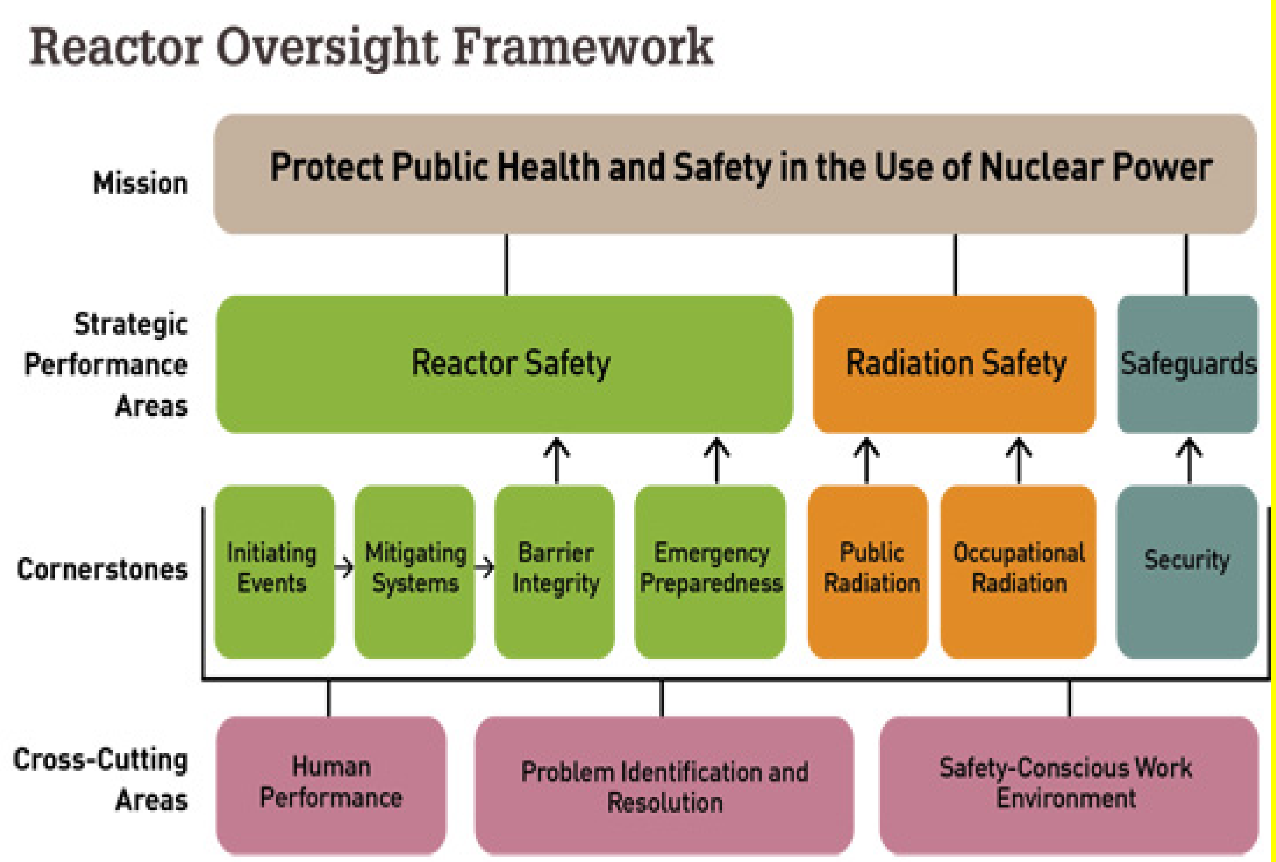
Vision and Strategy for ARCOP

New and innovative thinking and consequence and safety-significance based approaches are required for construction inspection and oversight of SMRs and advanced reactors

NRC will leverage knowledge and experience from internal and external sources to inform and develop construction inspection and oversight for SMRs and advanced reactors

Framework for Advanced Reactor Construction Inspection and Oversight

- Broad range of new and advanced reactor designs
- New technologies, materials, and manufacturing techniques
- Various reactor sizes (MWt - micros to larger reactors comparable to current operating plants)
- Scope to include non-LWRs, LWR SMRs, and fusion
- Current cROP and ROP frameworks could be leveraged for advanced reactors
- Establish meaningful performance metrics for new and advanced reactors



VISION

Advanced Reactor Inspection and Oversight framework considers:

- Flexibility, scalability, and adaptability to a wide range of advanced reactor designs and technologies
- Balance between off-site manufacturing and on-site construction
- Use of risk-informed, performance-based licensing process
- Leveraging existing cROP and ROP frameworks where appropriate
- Lessons learned from fuel cycle facilities, RTRs, Moly-99, and new reactors
- Use of inspection, monitoring, and compliance assurance technologies and techniques from other industries
- Smart, efficient use of internal NRC resources with supplemental external expertise
- Consequence and safety-significance based approach

Construction Inspection and Oversight of Advanced Reactors

Expectations and Considerations:

- Initial focus on ARDP awarded technologies and microreactors to support near-term deployments (heat pipe, liquid metal-cooled fast reactor, and high temperature gas-cooled reactor)
- Consider various reactor sizes (from micro-reactors of 10’s of MWt to larger reactors of 100’s and 1000’s MWt)
- Scale up from RTRs - micro’s more like RTRs
- Transform and leverage traditional large-LWR approach
- Flexibility in approaches to developing an inspectable licensing basis (Part 50, Part 52, future Part 53)
- Leverage COVID-19 lessons learned and potential use of remote/virtual inspection capabilities
- DANU leads framework development based on experience with advanced reactor technologies and RTRs with transition to DRO and Regions
  - Coordinate with and leverage internal NRC expertise and experience
  - Supplement NRC experience with external expertise on non-LWR technologies, materials, fuels, and manufacturing techniques, as necessary

Proposed Plan and Long Term Vision

- Develop an ARCOP framework document that outlines an overall process that is technology neutral, risk-informed and performance based

- Technology inclusive scope includes non-LWRs, SMRs (i.e., LWRs less than 300 MWe) and fusion reactors

- Prioritize and focus development of individual inspection and oversight framework areas on near-term technology commitments - microreactors, liquid sodium-cooled and high temperature gas cooled reactors

- Inform development of overarching ARCOP program with lessons-learned from development and implementation of near-term technology-specific inspection and oversight plans

NRR Lead Team:
Eric Oesterle (DANU)
Joe Sebrosky (DANU)
Maryam Khan(DANU)
Bill Reckley (DANU)
Phil O'Bryan (DANU)
Arlon Costa (DANU)

NRR Subject Matter Experts:
DRO/IQVB - Vendor Inspection Branch
DRO/IRIB - Reactor Oversight Branch
VPO – Vogtle Project Office
DNRL/NRLB – Small Modular LWR Reactors
DNRL/Senior Technical Advisor – Advanced Additive Mfg.
Regions II Division of Construction Oversight
NSIR – Security and Emergency Preparedness

Next steps:

- NRC effort being supported by external contractor with subject matter experts in construction inspection, operational oversight, advanced reactor fuels and technologies
- Draft of framework to be developed over next 6 – 9 months
- Status of ARCOP efforts will be periodically communicated at stakeholder meetings
- Considering separate public meeting(s) on ARCOP effort for focused outreach and stakeholder feedback

Questions or Comments?

Advanced Reactors Construction Inspection and Oversight Framework

May 27, 2021

Advanced Reactor Exports Working Group

Lauren Mayros

International Policy Analyst
Export Controls and Nonproliferation Branch
Office of International Programs

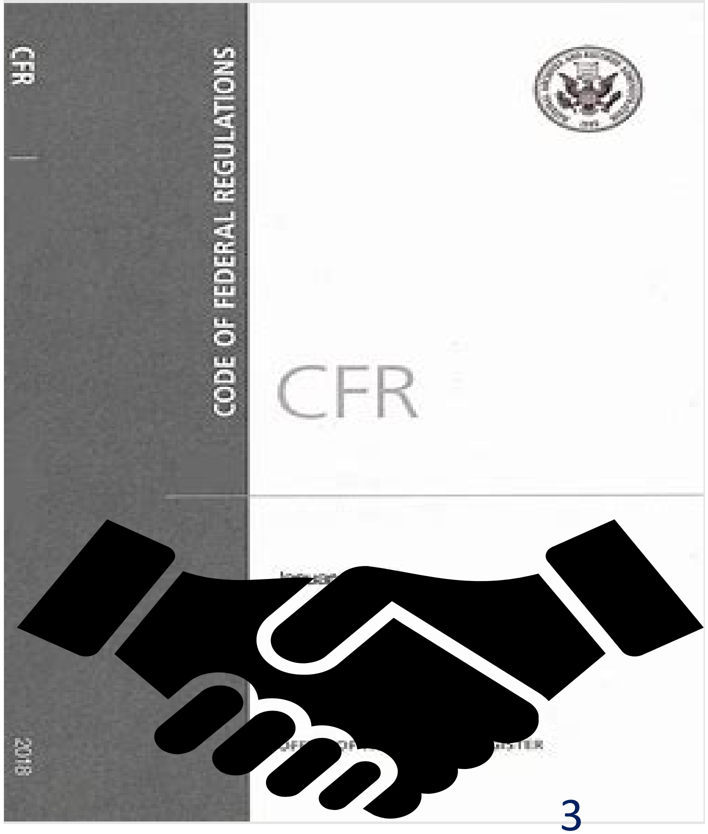
May 27, 2021

AREWG Purpose and Background

- Forward looking in the spirit of innovation and transformation.
- Keep pace with fast moving developments in the field of advanced reactors.
- Ensure that the NRC is prepared to license the export of these technologies in an independent, predictable and efficient way.

AREWG Mandate

- Evaluate NRC's readiness to complete exports (10 CFR 110) of "advanced reactors" to other countries consistent with NRC's Principles of Good Regulation (independence, openness, efficiency, clarity, and reliability).
- Assess if current level of review for advanced reactors is still appropriate.
- Conduct outreach to prospective vendors of advanced reactors on NRC's export licensing process.
- Develop a communication plan for future outreach.



Participants

- OIP
- OGC
- NMSS
- NSIR
- NRR
- RES
- Department of Energy/National Nuclear Security Administration
- Argonne National Laboratory

Design Types Studied

1) high temperature gas-cooled reactors

2) sodium fast reactors

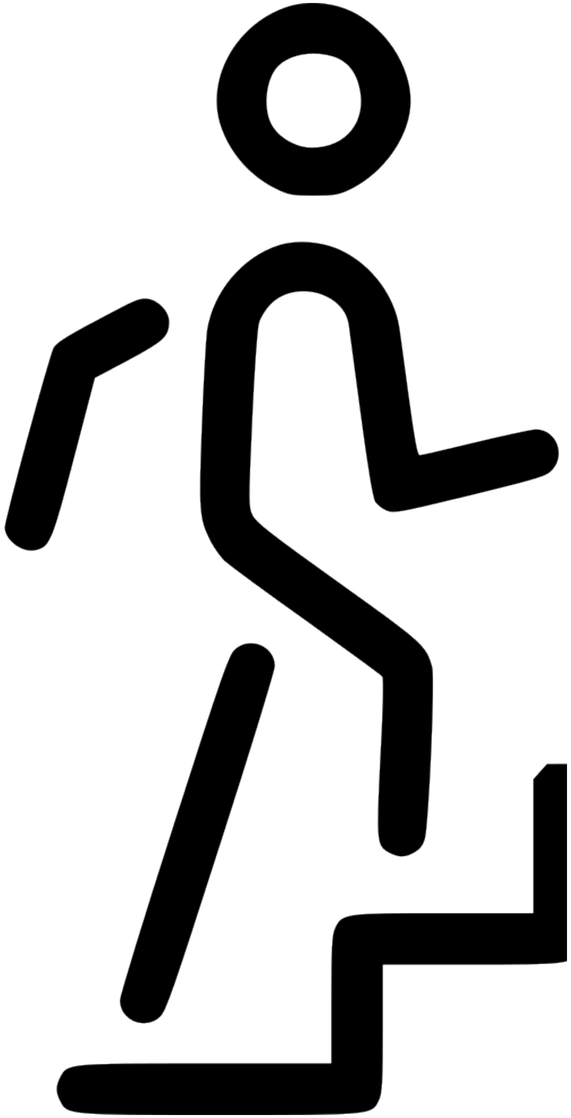
3) fluoride salt-cooled high temperature reactors

4) molten salt reactors, including liquid fluoride salt and liquid chloride salt-cooled reactors

5) small heat pipe reactors.

Conclusions and Recommendations

1. 10 CFR Part 110 is generally ready to license the materials and components associated with the 5 types of advanced reactor types studied.
2. Identified one advanced reactor system that is not clearly captured under Part 110 for export – the use of salt as a coolant.
3. Recommended several clarifying changes to Part 110 to remove any ambiguity that advanced reactors are covered under Part 110, i.e. fuel cladding other than Zirc. Tubes and salt.
4. Recommended working with the USG interagency to coordinate the recommended changes to Part 110 with the technical agenda of the NSG and conduct industry outreach on its conclusions.
5. Did not recommend changing the level of review for applications involving material and/or components for advanced reactors, i.e. Commission level review.



Next Steps

We want to hear from you!!

- We want to garner industry input as to whether a rulemaking or a reg guide would be the preferred way forward to clarify the provisions for advanced reactor exports under Part 110.
- Look out for the AREWG Public Report! Coming Soon to our website.

Questions

Thank You!

Any questions?

Break

Meeting will resume at 1pm EST

Microsoft Teams Meeting
Bridgeline: 301-576-2978
Conference ID: 550 337 464#

Graded Probabilistic Risk Assessment (PRA) to Support Advanced Reactor Licensing

Nathan Sanfilippo, Special Assistant &
Martin Stutzke, Senior Level Advisor for PRA
Division of Advanced Reactors and Non-Power Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Problem Statement

- Preliminary Part 53 rule text for discussion currently would require applicants to perform a probabilistic risk assessment (PRA) to support the development of the safety analyses for an advanced reactor application.
- Some potential applicants are questioning the need for, and burden of, performing a PRA for designs that may have significantly lower power levels and source terms than large light-water reactors (LWRs).
- The NRC staff has committed to evaluate the possibility of grading the PRA.

Approach

- Develop viable options consistent with the preliminary text of 10 CFR 53.450. Working group will consist of 3 phases:
  - Phase 1 – Graded PRA Concept: Craft options and align on conceptual graded PRA approach
    - Goal: Summer 2021
  - Phase 2 – Graded PRA Guidance: Draft guidance on agreed-upon approach
    - Timeline to support Part 53 needs. Goal: Fall 2021
  - Phase 3 – PRA Alternatives: Consider acceptable alternatives to PRA for meeting risk assessment requirements
    - Begin following Phase 1 and parallel to Phase 2 to support Part 53 timeline

Working Definitions

- ***Graded PRA approach*** means a process that uses bounding, conservative, and/or qualitative assessments to establish a PRA's scope, level of detail, degree of plant representation, and/or level of peer review commensurate with the licensing stage (which dictates the level of detail and finality of the information used to develop the PRA) and how the PRA will be used in risk-informed decision-making.
- ***A Graded PRA*** is a PRA of appropriate degree of scope, level of detail, plant representation, and technical adequacy to support a specific advanced reactor licensing application.
  **Graded* should not imply that a design is not yet complete – acceptance of a graded PRA could only be considered if a design is well understood and conservatively modeled.
- ***A Dose/consequence-based criterion*** is a potential entry condition to enable a graded PRA that uses bounding, conservative, and/or qualitative assessments of the doses or consequences arising from potential unplanned release scenarios, without consideration of the release scenario likelihood. This approach is being considered as a specific criterion for developing a graded PRA to adequately demonstrate that an applicant meets the intent of the Commission's Severe Accident Policy in an efficient and effective manner.

Advanced Reactor Program

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Goals

- Identify what criteria would be used to determine how a PRA would be graded (e.g., criteria of a dose/consequence-based approach).
- Identify the purposes/applications for which the graded PRA can be used post-licensing based on its scope, level of detail, degree of plant representation, and/or level of peer review, and expected maintenance.
- Define the level of detail needed at different stages of the licensing process (e.g., what's needed at Construction Permit stage vs. Operating License stage).
- Consider how to ensure equivalent treatment of designs currently under review or soon to be received vs. what's in Part 53 in ~2025.

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Where We Started

(Slide 1 of 2)

- The Commission's advanced reactor policy statement (73 FR 60612; October 14, 2008) indicates the following:
  - Use PRA as a design tool, as implied by the Commission's PRA policy statement (60 FR 42622; August 16, 1995).
  - Use PRA to search for severe accident vulnerabilities, in accordance with the Commission's severe accident policy statement (50 FR 32138; August 8, 1985).
  - Comply with the Commission's safety goal policy statement (51 FR 28044; August 4, 1986, as corrected and republished at 51 FR 30028; August 21, 1986).

Where We Started

(Slide 2 of 2)

- The non-LWR PRA standard (ASME/ANS RA-S-1.4-2021) was developed to support PRAs performed in various stages of design and licensing. From Section 1.2: "...the requirements in this Standard for the level of detail, completeness, and model to plant or design fidelity vary according to the scope and level of detail of design and operational information that is available to support, and is referenced by, the PRA with additional requirements to address assumptions in lieu of as-operated and as-built details."

Current Uses of PRA

(Slide 1 of 2)

- Identify severe accident vulnerabilities and provide insights which, if addressed, support the conclusion that the plant design, construction, and operation provides reasonable assurance of no undue risk to public health and safety.
- Demonstrate that the plant meets the Commission's safety goals.
- Support the environmental review required by 10 CFR Part 51, specifically the evaluation of Severe Accident Mitigation Design Alternatives (SAMDAs) (see RG 4.2 and COL-ISG-029).
- Select licensing basis events (LBEs), classify structures, systems, and components (SSCs), and inform the defense-in-depth adequacy evaluation (for applications based on the Licensing Modernization Project (LMP) guidance).
- Support the process used to demonstrate whether the regulatory treatment of non-safety systems (RTNSS) is sufficient and, if appropriate, identify the SSCs included in RTNSS (for applications not based on the LMP guidance).

Current Uses of PRA

(Slide 2 of 2)

- Identify and support the development of specifications and performance objectives for the plant design, construction, inspection, and operation, such as:
  - Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC);
  - Reliability assurance program;
  - Technical specifications; and
  - Combined License (COL) action items and interface requirements.
- Support various voluntary risk-informed applications (e.g., risk-informed in-service inspection) that may be included in the licensing application.
- Inform the scope of staff's review; see SRM-COMGBJ-10-0004/COMGEA-10-0001 (ML102510405).
- Support the Reactor Oversight Process (ROP).

Role of PRA

- For Part 50/52 applications that are not based on the LMP guidance, PRA plays a *confirmatory/supporting* role in establishing the licensing basis.
- Part 50/52 applications that are based on the LMP guidance, PRA plays a *leading* role in establishing the licensing basis.
- For future Part 53 applications, PRA plays a *leading* role in establishing the licensing basis.

Where We're Going

- Looking for opportunities to use bounding, conservative, and/or qualitative assessments to establish a PRA's scope, level of detail, degree of plant representation, and/or level of peer review commensurate with how the PRA will be used in risk-informed decision-making.
- An individual criterion might be hard to identify in isolation, but perhaps a combination of criteria could be used to grade a PRA or accept an alternative approach that meets the Commission's expectations to assure safety.
- For large LWRs, PRAs were used to reduce the uncertainty involved with conservative deterministic designs leveraging the benefit of years of operating experience and data. For non-LWRs without deterministic design criteria and without comprehensive operating experience or test data, what is the appropriate approach to grading the PRA?
- Ensuring a comprehensive search for initiating events.

Discussion Topics

(Slide 1 of 3)

The NRC is interested in any feedback regarding the topic of Graded PRA, such as:

- What criteria should the NRC use to determine when a graded PRA may be performed?
  - Reactor thermal power;
  - Conservative deterministic calculations of accident doses show margin to regulatory limits; and/or
  - The design provides enhanced margins of safety and/or uses simplified, inherent, passive, or other innovative means to accomplish its safety and security functions (i.e., the design has one or more of the attributes identified in the Commission's advanced reactor policy statement).
  - Other criteria or considerations?

Discussion Topics

(Slide 2 of 3)

- Are there specific ways that applicants would envision the scope, level of detail, and/or degree of plant representation of a PRA be reduced?
- What are the advantages and disadvantages of reducing the PRA scope according to:
  - The radiological sources addressed by the PRA?
  - The plant operating states addressed by the PRA?
  - The hazard groups (internal initiating events, internal floods, internal fires, seismic hazard, high wind hazards, etc.) addressed by the PRA?

Discussion Topics

(Slide 3 of 3)

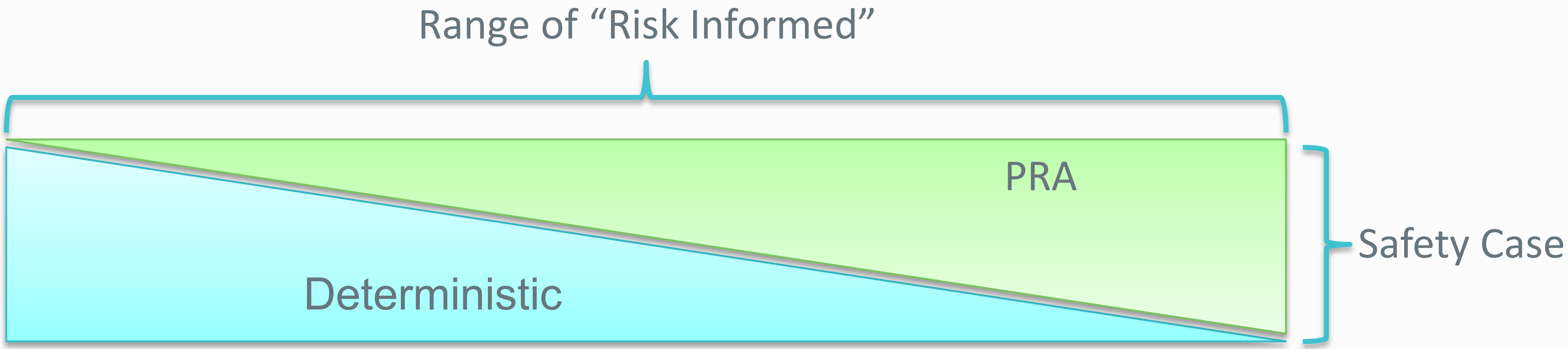
- The non-LWR PRA standard calls for the performance of a seismic PRA. For ALWRs, the NRC staff has endorsed in DC/COL-ISG-020 the use of PRA-based seismic margins analysis. Are there acceptable alternatives for assessing seismic risk?
- Are there alternatives to PRA that accomplish the same Commission objectives?

Part 53
Graded Approach to PRA

May 27, 2021

NUCLEAR ENERGY INSTITUTE

Graded Approach to Risk-Informed Licensing Basis



- Benefits of Deterministic
  - Fewer resources to develop
  - Bounding assessments encompass uncertainties
  - Margin typically not quantified
- Benefits of PRA
  - Prioritize a broader set of potential challenges to safety
  - Provides insight into margin
  - Operational flexibility by focusing on important SSCs

Note: In some cases a qualitative risk evaluation might be an acceptable substitution for a PRA. This can be considered after establishing the range of risk-informed approaches acceptable under Part 53.

Range of Licensing Basis Approaches Requires Flexibility in the Role of the PRA

Attribute	Minimal PRA Role	Focused	Maximal PRA Role
Best Situation for Use	Simple designs, very low consequences/high safety margins	Prefer to use PRA insights to inform design, designs are more complex	Prefer to maximize benefits of PRA (design and operational)
Licensing Approach Examples	Maximum Credible Accident	Traditional, IAEA	LMP+TICAP
Role of PRA in Safety Case	Risk insights validate selection of maximum credible accident	Identify and address potential design vulnerabilities, compare to Safety Goal Policy	Event selection, SSC classification, DID, margins, QHO
Role of deterministic methods	Event selection, classification, DID, safety analysis	Event selection, classification, DID, safety analysis	Safety analysis of DBAs
Scope/Level of PRA Detail in Application	Description that PRA validates MCA selection, PRA available for Audit	Description of PRA and results, PRA available for Audit	Per TICAP/ARCAP
Regulatory Controls/ Special Treatments	Only those necessary to assure MCA	Determined by PRA, necessary mitigations	Per LMP

Increasing Reliance on PRA

How the PRA is performed is derived from it’s role in the licensing basis

Attribute	Minimal PRA Role	Focused	Maximal PRA Role
Radiological Sources	Fueled reactor only	As appropriate to role of PRA	All
Plant Operating States	Maximum credible only	As appropriate to role of PRA	All
Hazard Groups	Maximum credible only	As appropriate to role of PRA	All
Types of PRA Methods	As necessary to confirm reasonableness of MCA	ANLWR PRA Standard and alternatives as appropriate	ANLWR PRA Standard
Treatment of External Hazards	Design essential functions to current external standards	Design essential functions and mitigations to current external standards	Per RG 1.233/NEI 18-04
BDBE	Included as MCA	Mitigation strategies	QHO + Mitigation
Regulatory Controls/ Special Treatments	Only those necessary to address MCA	Determined by PRA, necessary mitigations	Per LMP

Increasing Reliance on PRA

Advanced Reactor Content of Application "Risk-Informed Technical Specifications" Interim Staff Guidance

May 27, 2021
Periodic Advanced Reactor Stakeholder Meeting

Overview: TICAP / ARCAP

- Technology Inclusive Content of Application Project (TICAP)
  - Scope is governed by the Licensing Modernization Project (LMP)-based safety analysis report
  - LMP process uses risk-informed, performance-based approach to select licensing basis events, categorize structures, systems, and components (SSCs) and ensures defense-in-depth (DID) is considered
  - Industry developing key portions of TICAP guidance – does not include guidance for technical specifications (TS)

- Advanced Reactor Content of Application Project (ARCAP)
  - Purpose is to develop technology-inclusive, risk-informed and performance-based application guidance
  - Being developed to support Title 10 of the Code of Federal Regulation (10 CFR) Part 50, Part 52, and Part 53 applications
    - Near-term need to develop guidance to support expected advanced reactor Part 50/52 applications using the LMP process endorsed in Regulatory Guide (RG) 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors"
    - Guidance will be updated as Part 53 rulemaking language is adjusted
  - Encompasses and supplements TICAP including guidance for TS

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Risk-Informed Technical Specifications: Background

- Section 182a of the Atomic Energy Act of 1954, as amended, requires applicants for nuclear power plant operating licenses to provide TS…such TS shall be a part of any license issued.
- In 10 CFR 50.36, "Technical Specifications," the Commission established its regulatory requirements related to the content of TS.
- Pursuant to 10 CFR 50.36, TS for operating nuclear power reactors are required to include items in the following categories: (1) safety limits and limiting safety system settings (LSSS), (2) limiting conditions for operation (LCOs), (3) surveillance requirements, (4) design features, and (5) administrative controls.
- The latest large light water reactors (LWR) applications used the standard TS NUREGs as guidance (e.g., NUREG-1431, Volume 1, "Standard Technical Specifications – Westinghouse Plants").

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Risk-Informed Technical Specifications ISG: Applicability

- This interim staff guidance (ISG) is applicable to applicants for non-LWRs, stationary micro reactors, and small modular LWRs submitting risk-informed applications for a construction permit (CP)* or operating license (OL) under 10 CFR Part 50 or for a combined license (COL), design certification (DC), or manufacturing license (ML) under 10 CFR Part 52.

- Once the content of Part 53 is developed this ISG can be updated where necessary and will then also apply to applicants for a power reactor CP, OL, COL, DC, or ML under 10 CFR Part 53.

* *An applicant for a CP under 10 CFR Part 50 is required by 10 CFR 50.34(a)(5) to include in the preliminary safety analysis report (PSAR) an identification and justification for the selection of those variables, conditions, or other items which are determined to be probable subjects of TS for the facility, with special attention given to those items which may significantly influence the final design. As an option, a CP applicant may propose preliminary TS and include them in the PSAR or in a separate application document.*

Risk-Informed Technical Specifications ISG: Applicability

- The content of this ISG aligns with Regulatory Guide (RG) 1.233, which endorses Nuclear Energy Institute (NEI) 18-04, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development.”

- For risk-informed applications that do not use NEI 18-04 methodology, applicants should discuss with the NRC staff in pre-application interactions how their TS approach differs from that proposed in this ISG and addresses the underlying requirements of 10 CFR 50.36.*

*Specific guidance for non-LWR, non-LMP based applications is being deffered based on identified near-term needs and focused application resources.

Risk-Informed Technical Specifications ISG: Related Guidance

- Other NRC guidance referenced in this ISG:
  - RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” describes a general approach to risk-informed regulatory decision-making and discusses specific topics common to all risk-informed regulatory applications.
  - RG 1.177, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications.” While RG 1.177 is focused on methods acceptable to the NRC staff for assessing the use of risk analysis of proposed changes to TS, its guidance is useful in evaluating certain aspects of initial TS development.

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Risk-Informed Technical Specifications ISG: Guidance Approach

- The text in the 10 CFR 50.36 regulations for TS content require adaptation to correlate to the analysis and outputs of the risk-informed approach described in NEI 18-04.
  - 10 CFR 50.36 requirements for safety limits, LSSS and LCO Criteria 1 through 3 involve challenges to the "integrity of a fission product barrier."

- To evaluate the acceptability of risk-informed TS for advanced reactors, this ISG correlates the 10 CFR 50.36 text with appropriate NEI 18-04 process analysis/outputs. These analysis/outputs include:
  - required safety functions (RSFs)
  - safety-related (SR) SSCs
  - frequency-consequence (F-C) target
  - 10 CFR 50.34 dose limits

Yellow highlighting on subsequent slides identifies significant differences from the 10 CFR 50.36, LCO criteria text.

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Risk-Informed Technical Specifications ISG: TS Content

10 CFR 50.36(c)(2)	TS Content Based on Corresponding NEI 18-04 Output
Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility.	**Limiting conditions for operation** are the lowest functional capability or performance levels of equipment required for safe operation of the facility.
Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.	**Criterion 1**. Installed instrumentation that is used to detect, and indicate where necessary, a significant abnormal degradation of barriers necessary to maintain the release of radioactive materials from the plant to within the design basis events (DBE) **F-C Target** or to mitigate design basis accidents (DBAs) that only rely on the SR SSCs to meet the dose limits of **10 CFR 50.34**.
Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.	**Criterion 2**. A process variable, design feature, or operating restriction that is an initial condition of an anticipated operational occurrence (AOO) or DBE necessary to maintain consequences to within the **F-C Target** or to mitigate DBAs that only rely on the SR SSCs to meet the dose limits of **10 CFR 50.34**.
Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.	**Criterion 3**. A structure, system, or component that is part of the primary success path and which performs a **RSF** to mitigate the consequences of DBEs to within the **F-C Target** or to mitigate DBAs that only rely on the SR SSCs to meet the dose limits of **10 CFR 50.34**.

Risk-Informed Technical Specifications ISG: TS Content

LCO Criterion 4

- In the Supplementary Information provided in the NRC's 1995 revision to the 10 CFR 50.36 TS regulation [60 FR 36953] (which codified the "Final Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors"), the Commission correlates Criterion 4 to r**isk-significant** SSCs that are:

  - …intended to capture those constraints that probabilistic risk assessment or operating experience show to be significant to public health and safety…to ensure adequate protection of the public health and safety or that the addition of such constraints provides substantial additional protection to the public health and safety

Risk-Informed Technical Specifications ISG: TS Content

- The NEI 18-04 process identifies two groups of SSCs that are tied to the substantial additional protection of public safety but are not addressed by LCO Criteria 1 through 3 discussed earlier:

  - SR SSCs that perform RSFs to prevent the frequency of beyond design basis events (BDBEs) with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C Target.
  - Non-safety-related SSCs relied on to perform risk-significant functions.

Risk-Informed Technical Specifications ISG: TS Content

10 CFR 50.36(c)(2)	TS Content Based on Corresponding NEI 18-04 Output
Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.	**Criterion 4**. (a) The group of SR SSCs relied on to perform **RSFs** to prevent the frequency of BDBEs with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C Target. (b) The group of Non-Safety-Related with Special Treatment (NSRST) SSCs relied on to perform **risk-significant** functions. These risk-significant SSCs are those that perform functions that prevent or mitigate any LBE from exceeding the F-C Target or make significant contributions to the cumulative risk metrics selected for evaluating the total risk from all analyzed LBEs.

Risk-Informed Technical Specifications ISG: TS Content

- Note that LCO Criterion 4 for the corresponding NEI 18-04 output does not include NSRST SSCs that only perform functions required for DID.

- This position is supported by NRC position paper SECY-94-084,”Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems In Passive Plant Designs” which describes “availability controls” for RTNSS* SSCs that address DID functions.

- Thus, NSRST SSCs that perform DID functions would fall into the “availability controls” (i.e., non-TS control document) category.

* RTNSS policies were developed in the 1990s to impose requirements on non-safety related SSCs that performed risk significant or DID functions. These policies were developed to address evolutionary advanced LWR designs that relied solely on the passive safety systems to demonstrate compliance with the acceptance criteria of various design-basis transients and accidents, and where designers designated all or most active systems as non-safety systems.

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Risk-Informed Technical Specifications ISG: TS Content

- Other ISG Guidance - LCO Format
  - A description of the operable condition
  - The mode applicability
  - The actions that must be taken when the operable condition is not met including any required action and the associated completion time (CT). For determining various LCO CTs the risk impact should be evaluated using the probabilistic risk assessment (PRA) and DID analysis. The ISG refers to RG 1.177, Regulatory Position 2.3.4 for additional guidance in this area.*
  - A set of associated surveillance requirements.

**The ISG notes that RG 1.177 references the risk metrics of core damage frequency (CDF) and large early release frequency (LERF) based on LWRs as factors in determining CTs. Advanced reactor applicants should use other risk metrics, such as those described in NEI 18-04 for determining CTs.*

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Risk-Informed Technical Specifications ISG: TS Content

- Other ISG Guidance - Surveillance Requirements
  - Surveillance requirements should be determined through the development of the “Special Treatments Considered for Programmatic DID” task in the LMP process.
  - The PRA and DID adequacy evaluations should provide a basis for determining the specified TS surveillance frequency.
  - Refer to RG 1.177, Regulatory Position 2.3.4 for additional guidance in this area.

Risk-Informed Technical Specifications ISG: TS Content

- Other ISG Guidance – Design Features
  - Similar to 10 CFR 50.36(c)(4) – “Design features affect aspects of the facility (e.g., construction materials and geometric arrangements) not covered in the categories described above that, if altered or modified, would have significant effects on safety.”
  - This requirement can again be correlated to the NEI 18-04 outputs for RSFs.

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Risk-Informed Technical Specifications ISG: TS Content

- Other ISG Guidance - Administrative Controls
  - Administrative controls are the provisions relating to organization and management, procedures, record keeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.
  - Administrative controls can be derived, in part, from the development of special treatments and the “Application of Programmatic DID Guidelines” described in the NEI 18-04 process.
  - ISG guidance in this area follows the latest standard TS NUREG guidance.

Risk-Informed Technical Specifications ISG: TS Content

- Other ISG Guidance - TS Bases
  - Similar to existing TS Bases
  - As an alternative, an applicant may provide the appropriate TS bases within the scope of the safety analysis report and alleviate the need to provide a separate TS Bases document. If this approach is used, the safety analysis report bases should clearly address each TS, other than those covering administrative controls.

Risk-Informed Technical Specifications ISG: TS Content

- Other ISG Guidance - Other Miscellaneous TS Content
  - A set of definitions for terms used in the TS
  - A definition of plant modes used in determining LCO applicability
  - A description of logical connectors (if used)
  - A description of the Completion Time conventions used in the TS and guidance for their use
  - A description of the proper use and application of surveillance requirement frequency requirements
  - An explanation of LCO applicability and what actions are necessary when an LCO is not met and associated Required Actions are not met

Risk-Informed Technical Specifications ISG

Comments/Questions?

Future Meeting Planning

2021 Upcoming Advanced Reactor Meetings (Tentative)
June 10, 2021 (Part 53 Public Workshop)
June 24, 2021 (Part 53 ACRS Subcommittee)
June 29, 2021 (SCALE/MELCOR Source Term Public Workshop – Heat-Pipe Reactor)
July 15, 2021 (Periodic Stakeholder Meeting)
July 20, 2021 (SCALE/MELCOR Source Term Public Workshop - HTGR)
September 14, 2021 (SCALE/MELCOR Source Term Public Workshop – Pebble-Bed Molten-Salt-Cooled Reactor)

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