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Alternate Safety Basis Methodologies for DOE Research Reactors

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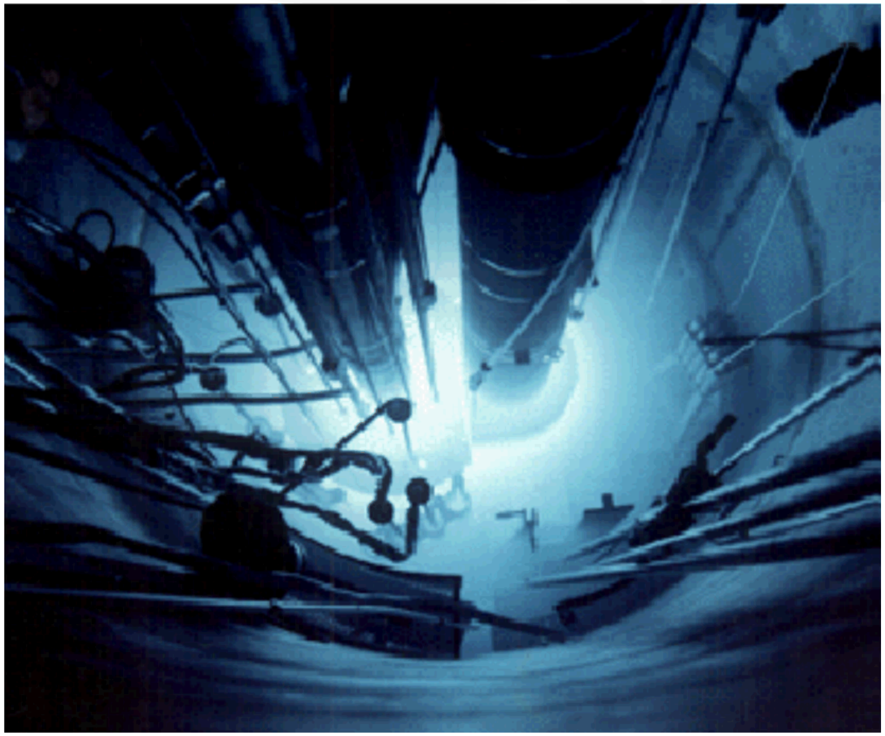
TA-V
LEADING NUCLEAR TECHNOLOGIES

ACRRF – Annular Core Research Reactor Facility

- TRIGA type research reactor
- Steady state, pulsed, and tailored transient modes
- 9” diameter central experimentation chamber (20” FREC-II diameter)
- ACRR core, with UO_2-BeO with 35% U-235
- FREC-II core, UZrH fuel with 20% U-235

Modes	Peak Power
Steady State	Up to 4 megawatt (MW)
Pulse	Up to 60,000 MW



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TRIGA Reactors in United States and World

- As of 2005, there were 19 operating in the US and 25 in the world excluding the US.
  - University of California-Davis, 2.3 MW,
  - Oregon State University, 1.1 MW,
  - Multiple US universities at ~ 1 MW (Pennsylvania State, Washington State, Texas A&M, University of Wisconsin)
  - **Institute for Nuclear Research, Pitesti, Romania, 14 MW**
  - Atomic Energy Research Establishment, Dhaka, Bangladesh, 3 MW
  - **Ongkharak Nuclear Research Centre, Bangkok, Thailand 10 MW**
  - Philippine Atomic Energy Commission, Quezon City, Philippines, 3 MW

Reference: University of Utah Nuclear Research Facility, License No. R-126, Docket No. 50-407, License Renewal Application, Updated Safety Analysis Report, June 2011, Redacted Version.

ACRRF Current Safety Basis Methodology

- ACRRF – Hazard Category 2 Reactor Facility
  - Safety Basis
    - Documented Safety Analysis (DSA) written to RG 1.70 and DOE-STD-3009 hybrid
    - Technical Safety Requirements (TSRs)
    - Safety Evaluation Report (SER) issued by SFO
  - Safety Basis is currently being upgraded as a result of new DOE guidance and DNFSB concerns (ACRRF Improvement Project).
  - Alternative Safety Basis methodologies are being explored

What is the Problem?

- Staff visits at Sandia TA-V nuclear facilities in 2011 – Reviewed the safety basis, instrumentation and control systems, and quality assurance program.
- Subsequent staff issued reports and letters sent to DOE concerning, in part, the adequacy of the safe harbor methodology used for the ACRRF safety basis.
- During this review, the Board's staff determined that NUREG-1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors* (1996), may be a more appropriate safe harbor for test reactors such as the ACRR. NRC regulators use NUREG-1537 for licensing of new non-power reactors.

Alternative Safety Basis Methodologies

- 10 CFR 830, Subpart B, Nuclear Safety Management
  - Provides "safe harbor" methodologies to develop Documented Safety Analyses for nuclear facilities. (Table 2 of 10CFR830, Subpart B)
  - If a contractor uses a method other than a safe harbor method, the contractor must obtain **DOE approval** of the method before developing the Documented Safety Analysis.

Potential Safety Basis Methodologies for Reactor Facilities at Sandia National Laboratories, TA-V

- **Regulatory Guide 1.70**, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants*, or successor document.
- **DOE-STD-3009** Change Notice 1, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, or successor document.
- **NUREG-1537, Part I**, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors* (1996)
- **ANSI/ANS 15.21-2012**, *Format and Content for Safety Analysis Reports for Research Reactors*
- Some combination of the above. (Possibly develop new standard for Safety Basis Methodologies for DOE Research Reactors)

Do Research Reactors Really Need Power Plant Controls?

- ACRRF does not require an emergency core cooling system because the reactor shuts down upon loss of coolant.
- ACRRF also does not require power plant controls such as:

Main steam isolation valve, main steam isolation valve, leakage control system, nuclear steam supply system, containment isolation system, automatic depressurization system, residual heat removal system (natural convective cooling is adequate to dissipate the ACRR core afterheat).

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Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Revision 3, 1978

- Specific to nuclear power reactors.
- Does not provide direction for how to evaluate experiments and associated Material at Risk impacts/release.
- Provides direction for safety basis content on many systems that are needed for power reactors, but are not present for research reactors.
- Does not specify an Evaluation Guideline for public consequences.
- Table 15-1 of RG 1.70 provides list of accidents that should be evaluated.

DOE-STD-3009 Change Notice 3 (successor to CN1)

- Specific to non-reactor nuclear facilities.
- Provides widely used methodologies for hazards evaluation and accident analysis.
- Requires development of unique and bounding accidents.
- Provides Evaluation Guideline for public consequences and more of a qualitative framework for worker risk.

NUREG-1537, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Part I

- DOE stated in letter back to DNFSB that NUREG-1537 is not a successor document to Regulatory Guide 1.70, so it is not considered a safe harbor methodology per 10 CFR 830 Subpart B.
- Used by many university research reactors (Oregon State University, University of Texas, University of Utah, implied at Texas A&M University, partially for Kansas State)
- Section 13 of NUREG-1537 provides list of accidents that should be evaluated.
- Indicates that reactors without instrumented fuel are allowed to establish a power level that limits fuel cladding maximum temperature below safety limits (Appendix 14.1). (could assist with obsolescence issues in ACRR).
- **Establishes very restrictive consequence acceptance criteria (10 CFR 20 or 10 CFR 100).**

ANSI-15.21-2012, Format and Content for Safety Analysis Reports for Research Reactors

- Updated in 2012, so this standard is more up to date than circa 1970s and 1990s standards.
- Experiment hazards and controls are considered in this methodology.
- No clear evaluation guideline for consequences; generic reference to applicable radiological acceptance criteria.
- Provides for 7 types of accidents to be addressed, including loss of electrical power, insertion of reactivity, loss of flow, loss of coolant, erroneous handling or malfunction of equipment or components, special internal events, and external events.
- Does not specify the method of conducting external accident evaluations (i.e., earthquake, wind event).

Types of Accidents to Be Evaluated

	RG 1.70	NUREG-1537	ANSI-15.21	DOE-STD-3009
Bounding Accidents			X	X
Unique Accidents				X
Maximum Hypothetical Accident		X		
Insertion of Excess Reactivity	X	X	X	
Loss of Coolant		X	X	
Loss of Coolant Flow	X	X	X	
Mishandling or Malfunction of Fuel		X	X	
Loss of Electric Power	X	X	X	
External Events		X	X	X
Mishandling or malfunction of experiment	X	X	X	
Other uncontrolled release of radioactive material	X	X		X

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Types of Accidents to Be Evaluated

	RG 1.70	NUREG-1537	ANSI-15.21	DOE-STD-3009
Increase in Heat Removal by the Secondary System	X			
Decrease in Heat Removal by the Secondary System	X			
Decrease in Reactor Coolant System Flow Rate	X			
Reactivity and Power Distribution Anomalies	X			
Increase in Reactor Coolant Inventory	X			
Decrease in Reactor Coolant Inventory	X			
Special Internal Events (internal fires, explosions, internal flooding, security incidents, experiment malfunctions)			X	X
External Man-Made Events (explosions from natural gas lines, aircraft crash)			X	X

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Alternate Methodology Recommendation

- Hybrid Approach using DOE-STD-3009 and ANSI-15.21-2012.
  - Overall structure of DSA/SAR content will be provided by following ANSI-15.21-2012.
  - DOE-STD-3009 will be used for hazards analysis methodology, consequence evaluation guideline, evaluation of natural phenomena hazards, and development of controls.
  - ANSI-15.21-2012 will be used for content associated with reactor parameters, safety management programs, pertinent systems and controls (i.e., cooling systems, instrumentation and controls), accident analysis and development of Technical Specifications.

Thank you!!! and Questions???

- Thank You! Safe travels home.

- Questions?

- Planned Future Work - Evaluate Safety Basis methodologies used by other research reactors in the DOE Complex.