ANS Standard 20.1
FHR Safety

FHR
Fluoride Salt Cooled High Temperature Reactor

"From the MSRE to a New Emerging Class of Reactors 50 Years Later"

Oak Ridge National Laboratory

Workshop on Molten Salt Reactor Technologies

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What is a Standard?

Taken from OMB Circular A-119: Federal Participation in the Development and Use of Voluntary Consensus Standards and in Conformity Assessment Activities

a. The term standard, or technical standard as cited in the Act includes all of the following:

(1) Common and repeated use of rules, conditions, guidelines or characteristics for products or related processes and production methods, and related management systems practices.

(2) The definition of terms; classification of components; delineation of procedures; specification of dimensions, materials, performance, designs, or operations; measurement of quality and quantity in describing materials, processes, products, systems, services, or practices; test methods and sampling procedures; or descriptions of fit and measurements of size or strength.

b. The term "standard" does not include the following:

(1) Professional standards of personal conduct.

(2) Institutional codes of ethics.

What is an ANS/ANSI Standard?

- Since 1957, the ANS Standards Committee has been active in the development of industry standards
- All ANS Standards receive dual approval from the American National Standards Institute (ANSI) and are thus considered American National Standards
- Industry standards are often started to establish safe practices
  - Once a standard is written and approved, it may influence regulatory guidance
  - Standards developed under voluntary consensus procedures, like those of ANS, often receive wide acceptance in their industry due to the broad representation of experts who worked to create the standard
- Since its start, ANS has written and approved nearly 150 different standards, and numerous versions of many of those
- At present ANS has more than 70 standards that are current American National Standards and many others that are considered historical standards

*From ANS Standards Website

What is the ANS 20.1 FHR standard?

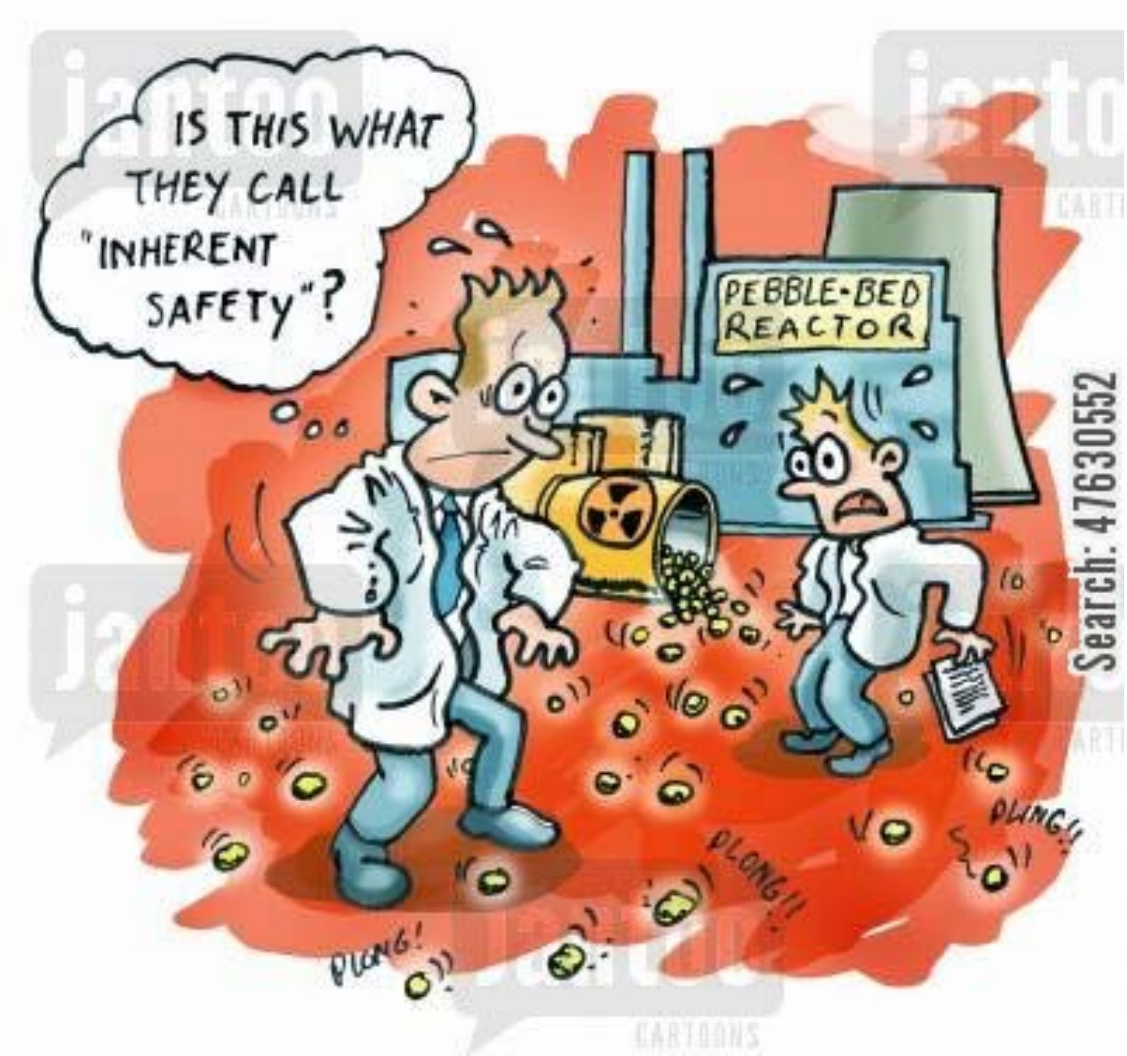
- ANS 20.1 – "*Nuclear Safety Criteria and Design Process for Fluoride salt-cooled High-temperature Reactor Nuclear Power Plants*"
- Standard charter:
  - This standard defines the safety objectives, general design criteria, and functional performance requirements for fluoride salt cooled high-temperature-reactor (FHR) nuclear power plants. The standard also describes the process for selection of licensing basis events for FHRs and classification for their systems, structures, and components (SSCs). The standard also defines requirements for the use of probabilistic risk information to establish performance-based, risk-informed criteria to inform reactor safety design decisions.

ANS 20.1 Working Group Members

- Co-chairs: Ed Blandford (UNM), Zhimin Dai (CAS), Matt Denman (SNL)
- Original working group: George Flanagan (Oak Ridge National Laboratory), Tom Kevern (United States Nuclear Regulatory Commission), Matt Memmott (Westinghouse), Charles Forsberg (Massachusetts Institute of Technology), Ron Cocherell (Southern Nuclear Company), Per Peterson (UC Berkeley), Zhaolin Chen (Chinese National Nuclear Safety Administration), Bojan Petrovich (Georgia Tech), Carl Stoots (Idaho National Laboratory)
- New membership: Andrew Sowder (EPRI), Jan Mazza (NRC)

There are other related advanced reactor standards

- ANS 53.1 – Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants
- ANS 54.1 - Nuclear Safety Criteria and Design Process for Sodium Fast Reactor Nuclear Power Plants
- New standard for salt-fueled reactors recently formed


IS THIS WHAT THEY CALL "INHERENT SAFETY"?
PEBBLE-BED REACTOR
PLONG!
PLONG!!
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Standard Uses Risk-Informed and Performance-Based Insights

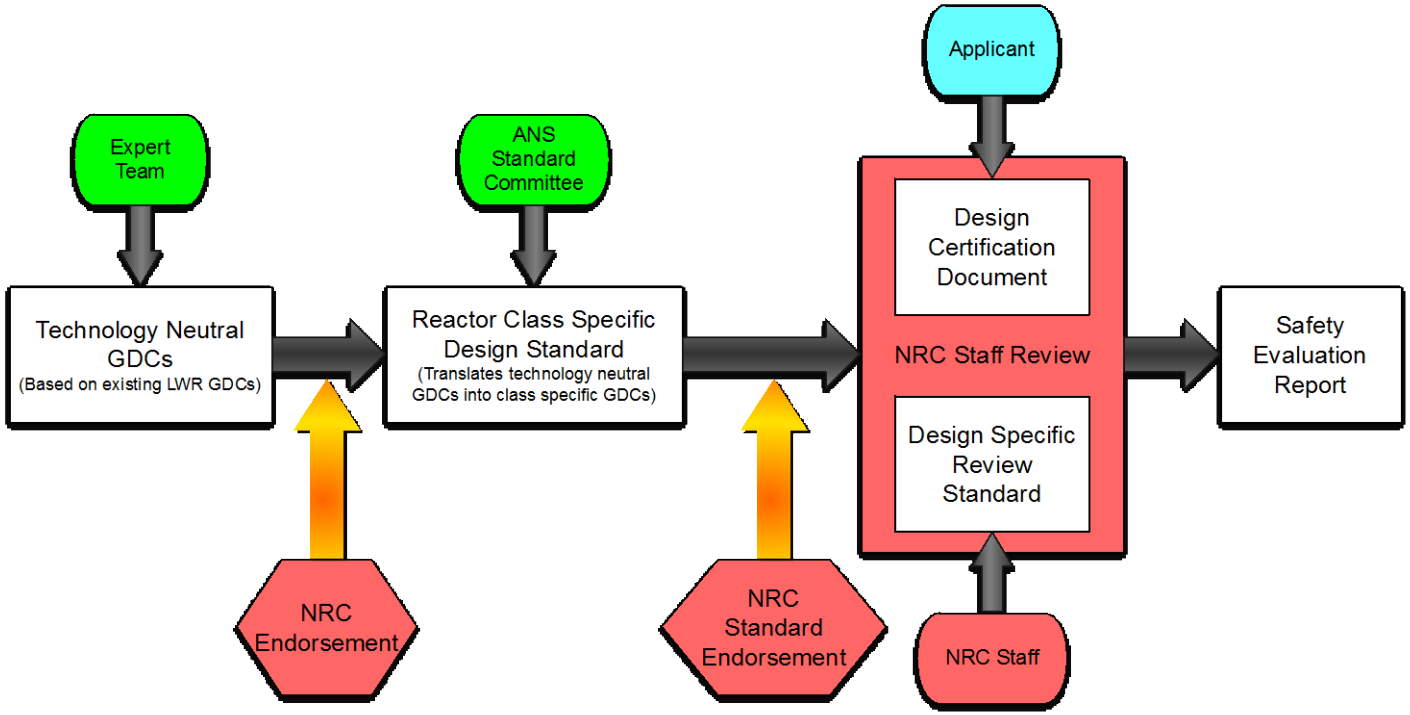
- Like 54.1, 20.1 allows for the option to use risk-informed and performance-based approaches
  - The Risk-informed and Performance-based Principles Policy Committee (RP3C) is keeping tabs on standards that utilize these methods
- What makes a standard performance-based?
  - A performance-based standard is one which focuses on attaining specific objectives
- What makes a standard risk-informed?
  - A risk-informed standard enhances the traditional, deterministic design approach by applying risk insights, through a deliberative process, to the plant design to focus resources on the most safety significant issues

20.1 Table of Contents

1. Introduction
   - Purpose
   - Scope
   - Overall approach
     - Traditional deterministic approach - (defense-in-depth (DID), safety margins, single-failure-criterion (SFC), design basis events (DBEs), design basis accidents (DBAs), general design criteria (GDCs))
     - Risk Informed Approach - (PRA, enhancements to DBAs, SFC, safety class, SA selection)
   - Overall design considerations (CSAU, qualified analysis tools, etc.)
2. Definitions
3. Deterministic
   - GDCs - groupings similar to 10CFR50 App. A
   - Initiating event identification and categorization
   - Transient and accident evaluation
   - SSC classification
4. Risk-Informed
   - PRA Quality / Scope
   - PRA Uses
     - Confirm plant safety
     - Assess for vulnerability (test DBA selection, test for balanced risk)
     - Enhance design (Safety classification, SFC, Special treatment [QA, EQ])
5. References

DOE & NRC have Initiated a Collaborative Effort on Technology Neutral Licensing of Advanced Reactors

- Developing modified set of GDCs for advanced (non LWR) reactor classes – (INL/EXT-14-31179)
- ANS 20.1 will provide class specific criteria
- Design specific review standard analogous to NUREG-0800 for LWRs will also be necessary



20.1 Can Fit Within SFR and HTGR Proposed ARDC Language*

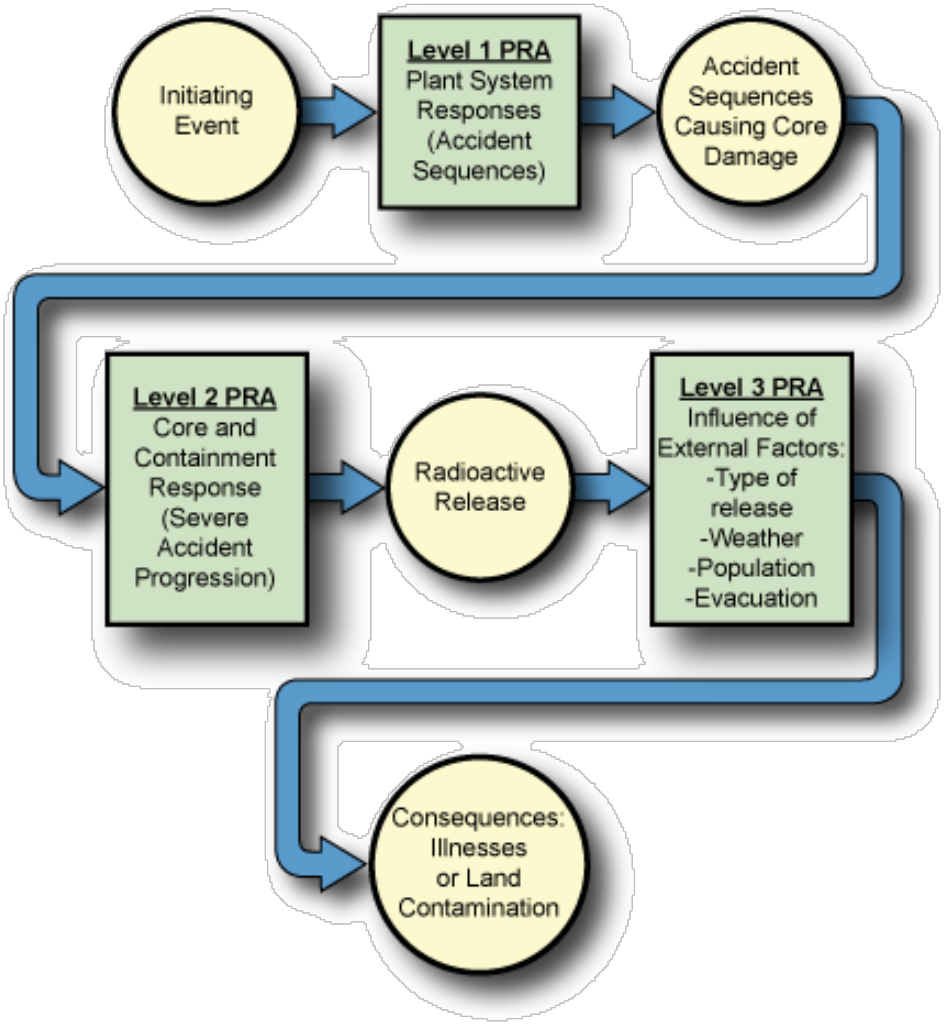
- A set of technology-neutral Advanced Reactor Design Criteria (ARDC) were proposed to the Staff in 2014
  - The proposed criteria were developed to preserve the underlying safety bases expressed by the original GDC, and recognizing that advanced reactors may take advantage of various new passive and inherent safety features different from those associated with LWRs
- 20.1 WG met in March to review ARDC language and to identify and necessary FHR-specific modifications

*One exception is ARDC 26 on *Reactivity Control Systems Redundancy and Capability* which we felt needed to be adjusted based on the relative high freezing temperature of the coolant to keep the original GDC intent

Next Steps

- Working group will be holding next meeting in November/December in order to finalize our first draft of the standard and ready for submittal in early 2016
- Welcome any interested parties and information sharing with new fluid-fueled MSR standard
  - Most meetings are held during ANS meetings and FHR-related workshops

Much Needed Work on FHR-Specific PRA



Potential Accident End States for the FHR

- Status of the primary FHR Primary Reactor Boundary (PRB)
  - Transients with intact PRB
  - PRB failures of different sizes and locations
  - Response of actions to isolate the PRB
  - Leak path through the PRB
- Status of the reactivity control systems
  - Sequences with successful automatic reactor trip
  - Sequences with manual reactor trip
  - Sequences with no reactivity control system response
- Status of the core heat removal systems
  - Sequences with continuous forced circulation cooling
  - Sequences with interruptions in forced circulation cooling
  - Sequences with conduction cooling

Potential Accident End States for the FHR

- Status of the protection against loss of coolant inventory
  - Sequences with no buffer salt reflood
  - Sequences with different degrees of buffer salt reflood
  - Sequences with different degrees of buffer salt reflood
- Status of the reactor building and associated Systems, Structures, and Components (SSC's)
  - Sequences with no release from the PRB into the reactor building
  - Sequences with successful or unsuccessful response of the filtration systems
  - Sequences with loss of RB structural integrity

Importance of peak metallic internal temp

- The fact that FHRs will have very large (hundreds of degrees) thermal margin to fuel damage has important implications
  - Power level for a given reactor core design will be established by criteria other than thermal limits on fuel damage
  - Potential limiting criteria for FHR thermal power include peak metallic component temperature during design basis accidents, reflector lifetime due to neutron dose limits, and peak fuel particle power
- Understanding the geometry of the FHR reactor system is central to predicting the reactor response (reactivity and heat removal) during transients and accidents

Metallic internal failure modes

- A key FHR design issue is that under normal operation key metallic components operate at temperatures where creep occurs and where time-dependent behavior must therefore be considered
- Development of new Division 5 for Section III of the ASME Boiler and Pressure Vessel (BPV) Code, that covers rules for the design, fabrication, inspection and testing of components for use in high-temperature nuclear reactors
- The candidate materials for FHR metallic structures include Alloy Alloy N, 316 stainless steel, Alloy 800H, and Alloy 617
  - All but Alloy N have been studied for NGNP application

Questions or comments?