U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Sufficiency Gaps Identified in EPRI Report 3002025288 on Enhanced Risk-Informed Categorization Methodology for Pressure Boundary Components

Office of Nuclear Reactor Regulation

April 30, 2024

Agenda

- Opening Remarks (DORL LT)
- Key Messages and High-Level Overview (M. Mitchell)
- Detailed Discussion of Staff Analysis to Supplement (NRC Staff)
  - NRC Basis
  - EPRI Response or Question
- Public Comment
- Closing Remarks

Opening Remarks

Key Messages and High-Level Overview

1. The NRC staff wants to discuss what is meant with, “a detailed technical basis,” why having one documented is essential to start a detailed review of the TR, and the path forward to getting to one.

2. An adequate, detailed technical basis is necessary to answer the “whys” of the proposed methodology.

3. Answering the “whys” of the proposed methodology is not only essential for the current review, but for the use of the methodology going forward.

NRC's Whys

- Why does this criterion exist in your methodology?

- Why is this criterion written exactly this way?

Detailed Discussion of Analysis Supplemental Items 2, 3, and 4

Supplemental Items 2, 3, and 4 have a similar theme that a detailed, specific technical basis is necessary to confirm criterion 1 through 10 (of the 14 criteria proposed in the EPRI methodology) will be adequate to provide appropriate categorization and that any **reduction in margin** will continue to provide reasonable assurance of adequate protection, while complying with 10 CFR 50.69.

A detailed technical basis is necessary to **efficiently** evaluate EPRI's proposed methodology will not **mis-categorize** pressure boundary components.

Sufficiency gaps on the EPRI TR and supplemental information are provided in further detail on the next slide.

Detailed Discussion of Analysis Items 2, 3, and 4

Basis a – During the public meeting on February 12, 2024, EPRI stated that risk insights were used to identify and develop the pre-determined Criteria 1-10 (of the 14 criteria provided) in the proposed methodology. An explanation of how the risk insights were used in developing the methodology is not apparent in the information submitted on the docket.

Basis b – Table 1 of the supplement contained information behind the criteria, but the descriptions were brief and lacked sufficient supportive arguments to expand on the rationale to the proposed criteria. Specific to predetermined criteria 1-10, the supplement did not provide information to address the NRC staff's concern to ensure the categorization of LSS SSCs via the proposed methodology are indeed LSS. The technical basis would need to provide sufficient justification to support the adequacy of the criteria. This includes plant-specific observations, operating experience, and the supporting logic used to develop the criteria.

Basis c – The alternatives approved to date are related to risk-informed in-service inspection (RI-ISI) programs. These criteria may be sufficient for RI-ISI programs. However, the staff cannot determine if these criteria are technically sufficient under 10 CFR 50.69 without providing information and logic to support their use. This gap is similar to the basis for denial of the methodology in Section 3.3.3 of the Safety Evaluation for Limerick Generating Station Units 1 and 2, "Notice of Issuance of Amendments Nos. 261 & 223 and Denial" (ADAMS Accession No. ML23094A179).

Detailed Discussion of Analysis – Item 1

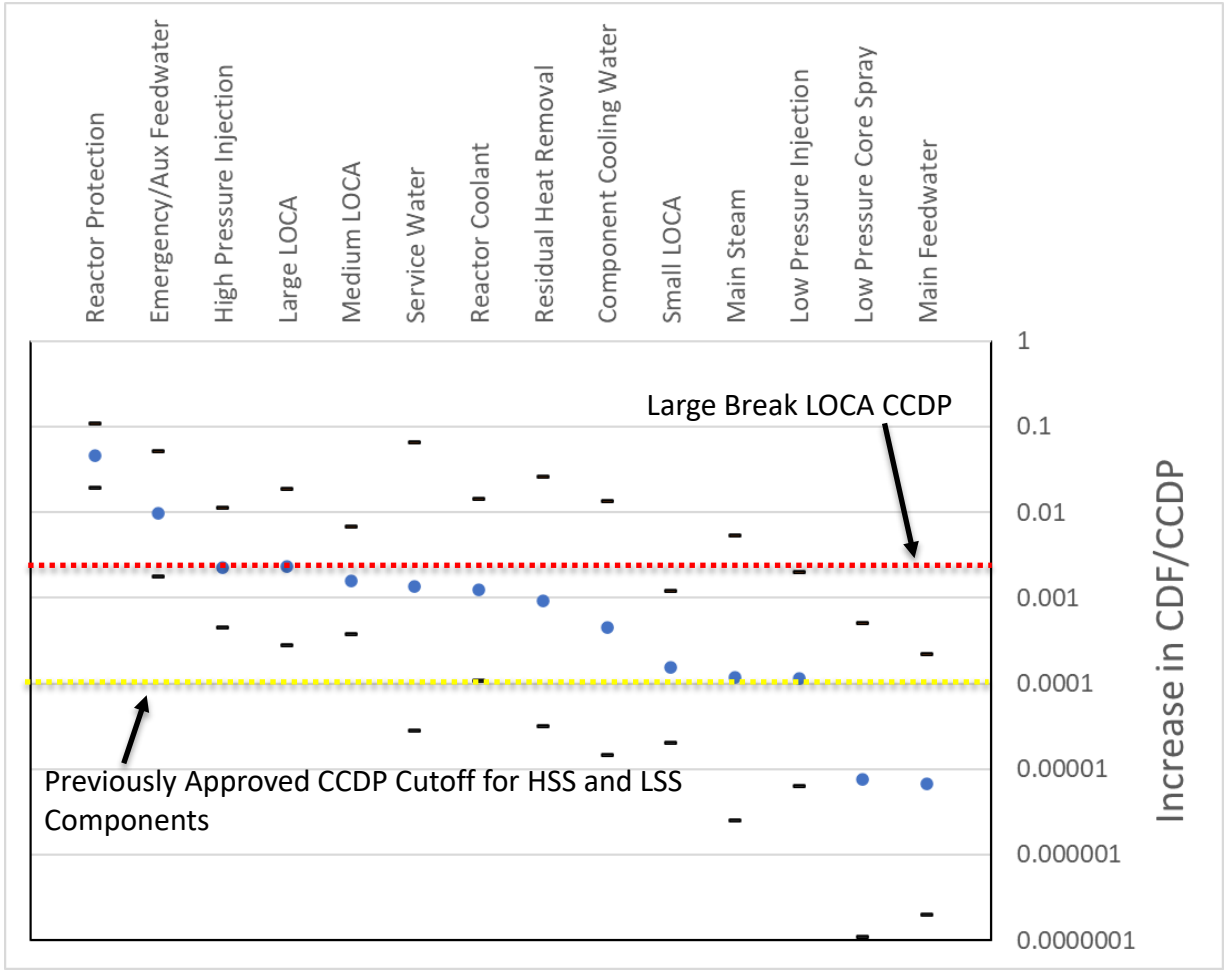
Item 1 – Recognizing that there are 14 criteria identified in the proposed EPRI methodology, generic criteria 1-10 have not been demonstrated to be applicable to the range of designs in the U.S. fleet.

Basis a – The question as to if the proposed criteria can be effectively applied to **different types and designs of plants** was not answered in either the original TR submittal or supplement.

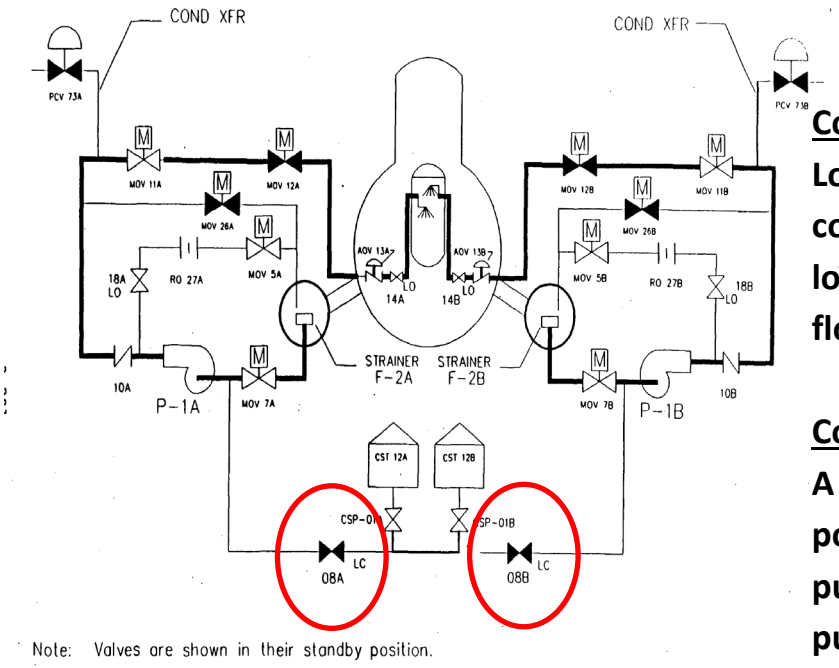
Basis b – Neither examples or technical bases were provided to demonstrate that all aspects of the integrated, systematic methodology will reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience, as required by 10 CFR 50.69(c)(1)(ii).

A few examples of SSCs that the NRC staff have identified for EPRI to consider using to demonstrate the implementation of the proposed categorization methodology are provided in the following slides.

Rankings from NRC PRA Database



EXAMPLE 1



Note: Valves are shown in their standby position.

Configuration:

Low Pressure Core Spray system in a BWR/4 (Mark I containment). Piping to pump suction from each CST to locked-closed valves 08A and 08B located in the grade level floor of a Reactor Building Crescent area.

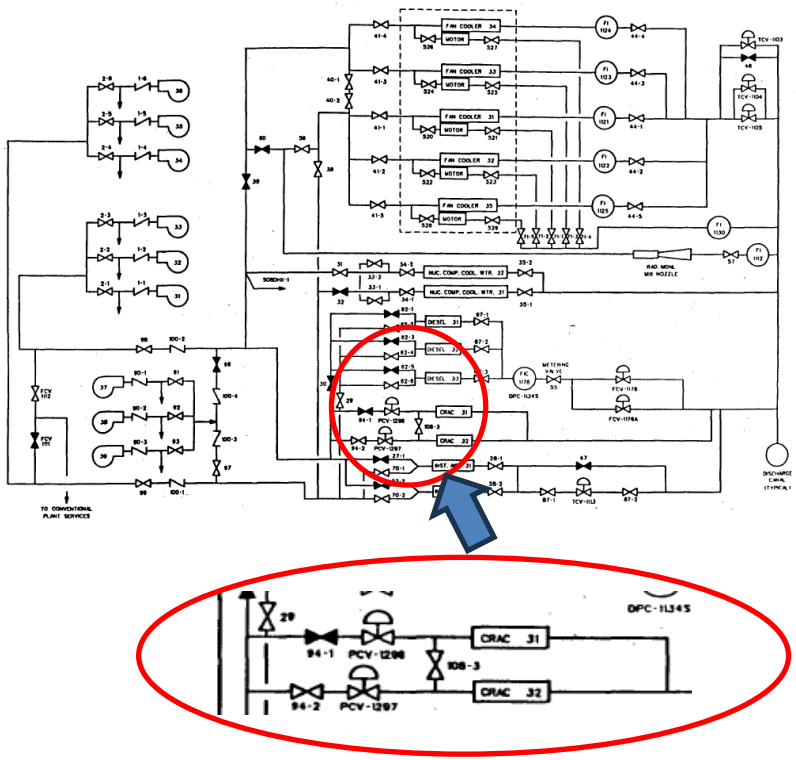
Concern:

A single rupture of either line in each Crescent Area has the potential to result in flood and spray damage to a core spray pump, LPCI/RHR pump on one loop, and either HPCI or RCIC pump controls. The scenario may be high safety significant because it could erode several functional areas for core cooling.

EPRI TR 3002025288:

This could be categorized as LSS since it falls outside the scope of Criterion 7 which only addresses failures associated with the CST for PWRs as HSS.

EXAMPLE 2



Configuration:

Common service water line in a 4-loop Westinghouse (pre-GDC) PWR from the Essential Header to both Control Room Air Conditioning (CRAC) unit condensers CRAC-31 and CRAC-32.

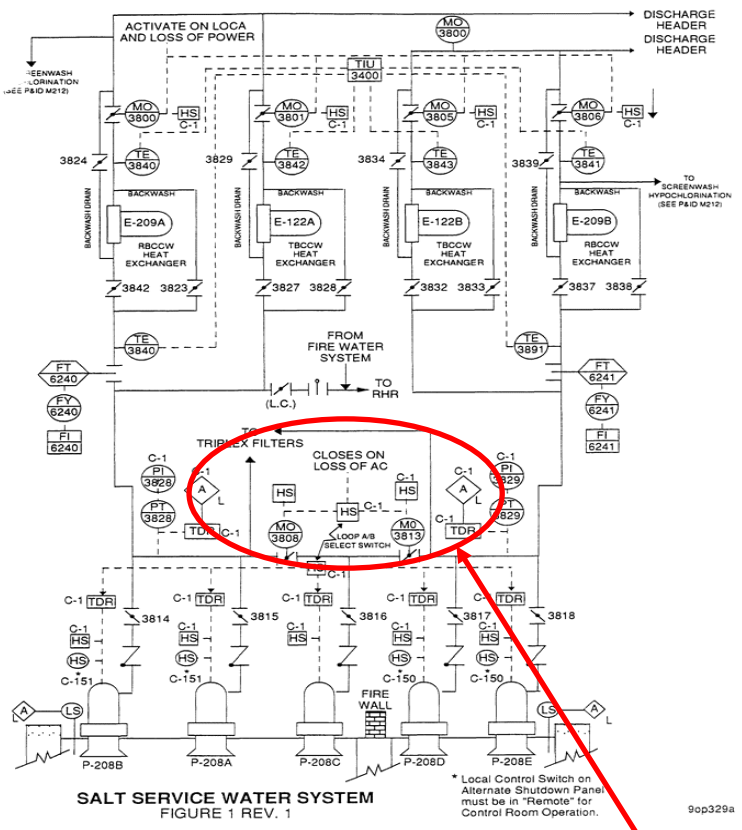
Concern:

A single rupture could lead to loss of control room air conditioning since both CRAC units are impacted, and therefore pose a control room habitability issue in an accident scenario requiring isolation of the control room.

EPRI TR 3002025288:

The staff is concerned that this would be categorized as LSS, per the EPRI proposed methodology, because the conditions of Criteria 1-10 may not be met. And, since control room HVAC is not modeled in most licensee model-of-records, there is no entry into Criteria 11-13 as a safety-net to determine as HSS.

EXAMPLE 3



Valves MO-3408 & MO-3813 are normally open and shut on loss of AC.

<u>Configuration</u>:

Salt Service Water system in a BWR/3 (Mark I containment). A single train system which will isolate to two trains on a LOOP or accident signal.

<u>Concern</u>:

A single rupture while both trains operating together could result in a catastrophic loss of cooling before isolation.

<u>EPRI TR 3002025288</u>:

Criterion 8 addresses this condition only for PWR plants.

Detailed Discussion of Analysis – Item 5

Item 5 - Operating experience shows that smaller SSCs fail at a higher rate than larger SSCs and that some smaller diameter piping with high-conditional core damage probability could be categorized as LSS.

Basis a – The supplemental information provided by EPRI for NRC Item 5 included the type of information that the staff is seeking. While this item might not rise to a non-accept issue by itself, it expands and supports the previous issue of lack of technical basis that the staff has identified. With the predetermined list for the entire NPP fleet, the staff needs additional information to understand how the methodology would be used for smaller diameter piping. Useful information could include CDF, LERF, CCDP, and practical effects of failure for the all Class 2 piping less than NPS 4 that would be designated as LSS using the proposed method for a variety of representative designs of PWR and BWR NPPs.

Detailed Discussion of Analysis – Item 6

Item 6 – Pipe rupture initiating event frequencies could change with the new treatments, such as: changes due to revised inspection, quality control, and repair/replacement activities. It is not apparent to the NRC staff that these changes are accounted for in the method.

The supplemental information provided by EPRI for NRC Item 6 includes sufficient information for the staff's acceptance review.

Public Comments/Question

Closing Remarks