Farley Nuclear Plant
Residual Heat Removal
Autoclosure Interlock Deletion
April 23, 2014

Farley Nuclear Plant RHR ACI Deletion
Agenda

- Introductions
- Purpose of Meeting
- Background
- Issue for Discussion
- Proposed Approach
- Summary and Conclusions

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Farley Nuclear Plant RHR ACI Deletion
Purpose of Meeting

Discuss the approach and obtain NRC feedback and expectations on the technical justification for the elimination of the Residual Heat Removal System (RHR) Autoclosure Interlock (ACI).

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Farley Nuclear Plant RHR ACI Deletion
Background

- RHR Autoclosure Interlock
  - Provides protection against overpressurizing the low pressure RHR system by the high pressure RCS
  - Ensures there is a double barrier between the RCS and RHR system when the plant is at normal operating conditions
  - Both RHR isolation valves close automatically if the pressure increases above the bistable setpoint
  - Helps prevent interfacing system LOCAs

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Background

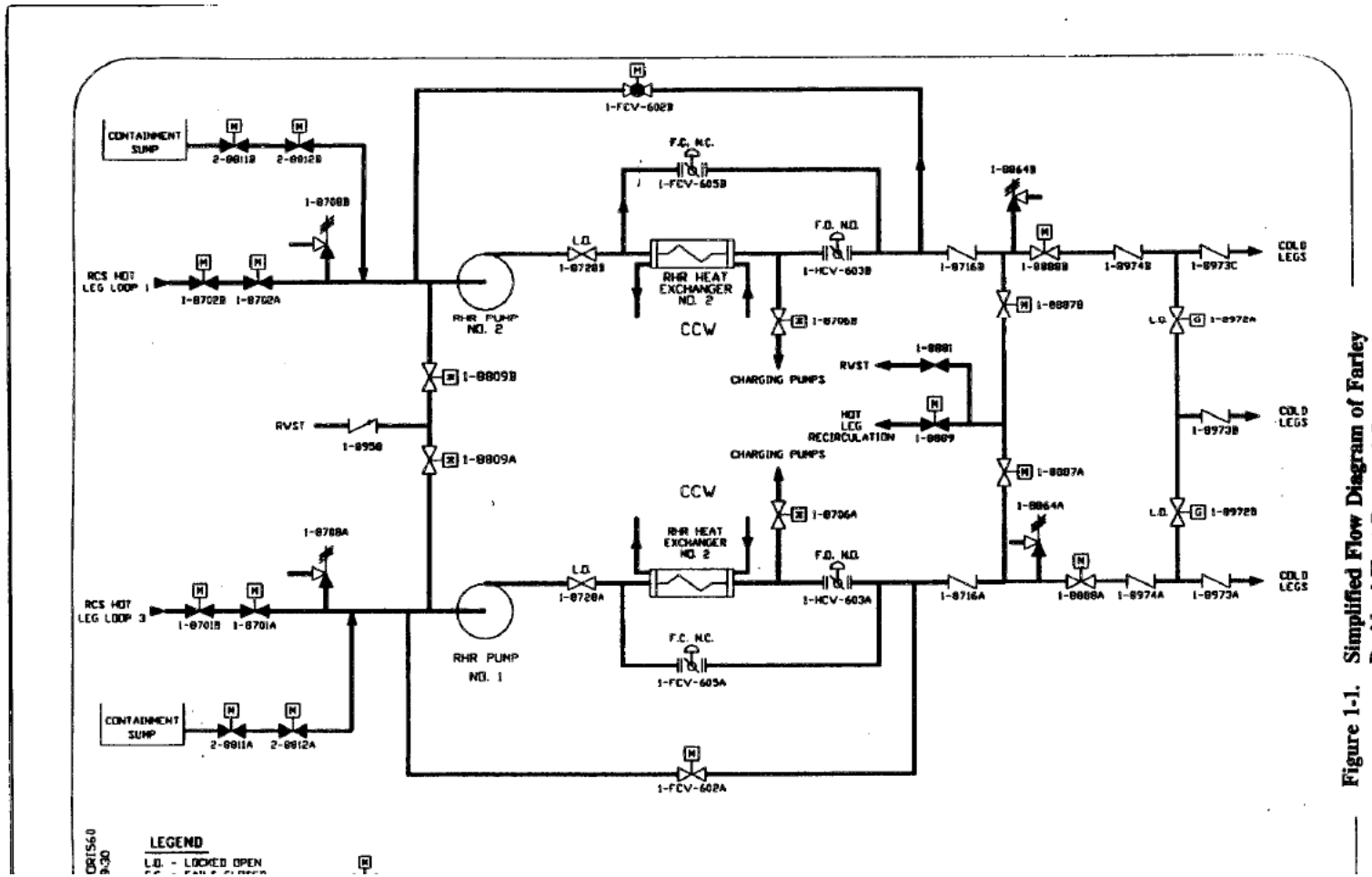

Figure 1-1. Simplified Flow Diagram of Farley Residual Heat Removal System

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Background

- Issue with RHR ACI
  - AEOD report “Decay Heat Removal Problems at U.S. Pressurized Water Reactors” dated December 1985, identifies 130 loss of RHR events in US PWRs between 1976 and 1983
  - 37 were caused by automatic closure of the RHR suction/isolation valves
  - Closure of the RHR valves results in a loss of cooling during shutdown (low pressure) operation

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Background

- NRC internal Memo on RHR ACI dated January 1985 stated:
  - A request to remove the ACI feature should be substantiated by proof that the change is a net improvement to safety, and
  - Should as a minimum address the following:
    - The means available to minimize Event V concerns.
    - The alarms to alert the operator of an improperly positioned RHRS MOV.
    - The RHRS relief valve capacity must be adequate.
    - Means other than the ACI to ensure both MOVs are closed (e.g., single switch actuating both valves)
    - Assurance that the function of the open permissive circuitry is not affected by the proposed change.
    - Assurance that MOV position indication will remain available in the control room, regardless of the proposed change.
    - An assessment of the proposed change's effect on RHRS reliability, as well as on LTOPs concerns.

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Background

- PWROG Program – RHR System ACI Removal
  - WCAP-11736-A, “Residual Heat Removal System Autoclosure Interlock Removal Report for the Westinghouse Owners Group”
  - Sorted Westinghouse NSSS plants into four groups
  - Provided assessments for four reference plants to demonstrate the change is a net improvement to safety
    - Interfacing system LOCA analysis
    - RHR unavailability analysis
    - LTOP/overpressurization analysis

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Background

- NRC SE on WCAP-11736-A: Staff Position
  - Removal of ACI for W NSSS plants can produce a net safety benefit provided five key improvements are in place
    - Alarms on RHR suction valves
    - Valve position indication
    - Procedural improvements
    - Power removed from RHR suction valves
    - Sizing of RHR valve operators
  - WCAP can be referenced in licensee's plant-specific submittals to show compliance with items that are not plant specific

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Background

- NRC SE on WCAP-11736-A
  - Section 2.4 – The effects of ACI removal upon plant safety must be evaluated on a plant-by-plant basis because of numerous plant-specific differences
  - Section 2.6 – The licensee should do sufficient PRA and safety analysis to ensure that its plant will not show results that will invalidate the conclusions of WCAP-11736
  - Requires submitting a LAR for the Tech Spec change
    - SR 3.4.14.2 in Tech Spec 3.4.14, “RCS PIV Leakage,” will be deleted

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Background

- Previous Farley RHR ACI Removal Program
  - WCAP-11746, Rev. 1, “Residual Heat Removal System Autoclosure Interlock Removal Report for the Joseph M. Farley Nuclear Plant Units 1 and 2” (April 1996) documents the justification for RHR ACI deletion at Farley
  - Addresses NRC’s requirements
  - Provides plant specific PRA for impact of RHR ACI removal on plant safety
    - Interfacing system LOCA analysis
    - RHR unavailability analysis
    - Low temperature overpressurization analysis

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Issue for Discussion

- The Farley specific analysis documented in WCAP-11746, Rev. 1 was completed in 1996
- The PRA analysis may not meet RG 1.200
- The analysis may not be acceptable to NRC Staff reviewers
- Potential issues with:
  - Fault trees
  - Data
  - Human reliability analysis
  - Event frequencies

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Proposed Approach

- Step 1: NRC meeting to discuss the proposed approach and obtain the Staff's feedback
- Step 2a: Assess the technical adequacy of the WCAP-11746, Rev. 1 models/analyses against ASME/ANS Standards and RG 1.200
  - Interfacing system LOCA analysis
  - RHR unavailability analysis
  - Low temperature overpressurization analysis

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Proposed Approach

- Step 2b: Categorize the deficiencies in meeting the PRA Standards
  1. Conservative aspect of the model that does not need to be addressed
  2. No impact on the decision-making process
  3. Could impact the results, but can be addressed via high level quantitative or qualitative assessment
  4. Impacts a key aspect of the analysis/models

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Proposed Approach

- Step 3: Model Changes and Quantification
  - Deficiencies will be addressed and these may be addressed with:
    - Model changes
    - Qualitative assessments
    - Sensitivity analyses
  - Data from most recent Farley PRA model will be used
  - All models will be re-quantified to demonstrate the acceptability of the RHR ACI deletion

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Farley Nuclear Plant RHR ACI Deletion
Summary and Conclusions

- Assess the technical adequacy of the WCAP-11746, Rev. 1 models/analyses against ASME/ANS Standards and RG 1.200
- Categorize the deficiencies in meeting the PRA Standards
- Model important changes and complete quantification
- Submit a LAR containing the above justification that supports the Tech Spec change
- NRC feedback on the proposed approach

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Questions or Comments?

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