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U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

LEVY NUCLEAR PLANT, UNITS 1 AND 2
DOCKET NOS. 52-029 AND 52-030
SUBMITTAL OF PRESENTATION MATERIALS FOR THE APRIL 7, 2016 ADVISORY COMMITTEE ON REACTOR SAFEGUARDS MEETING

Ladies and Gentlemen:

A meeting is scheduled for April 7, 2016 with the NRC's Advisory Committee on Reactor Safeguards to update the full committee on the status of AP1000 generic issues as they relate to Duke Energy Florida, LLC's (DEF) Levy Nuclear Plant Units 1 and 2 Combined License Application (COLA). The issues consist of Condensate Return, Main Control Room (MCR) Dose, MCR Heat Up, Hydrogen Venting and Protection and Safety Monitoring System (PMS) Flux Doubling. Enclosure 1 contains the presentation materials to be used by DEF and Westinghouse Electric Company LLC during the meeting.

If you have any further questions, or need additional information, please contact me at (704) 382-4046.

Sincerely,

Robert H. Kitchen
Director – Licensing
Nuclear Devleopment

Enclosure:
1) Levy COLA – ACRS Review AP1000 Generic Issues

cc : U.S. NRC Region II, Regional Administrator (w/o enclosure)
Mr. Donald Habib, U.S. NRC Project Manager (w/ enclosure)

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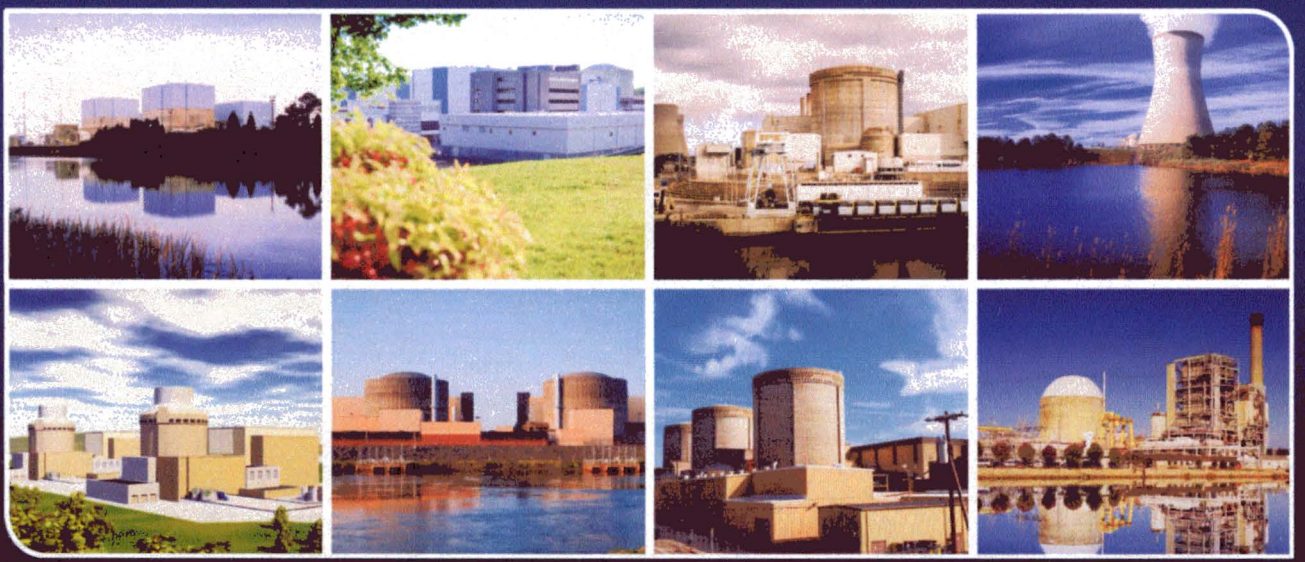
Enclosure 1

Levy COLA – ACRS Review
AP1000 Generic Issues
(18 pages including cover page)



Levy COLA – ACRS Review

AP1000 Generic Issues

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April 7th, 2016

Bob Kitchen – Duke Energy
Andy Pfister – Westinghouse

AP1000® PXS Condensate Return

Reason for the Design Change

- Previous analysis performed during design certification assumed a constant condensate return rate of 90%
- Investigations resulting from validation of this assumption determined the 90% return rate could not be met.
  - A result of as built design configurations that were different than testing used to establish the 90% return rate
- The safe shutdown temperature criteria in SECY-94-084 of 420°F in 36 hours could not be met with the calculated value of return rate without modifications.
  - Without the design enhancements, ADS actuation would have been sooner following a non-LOCA event. Adequate core cooling would have been maintained

Summary Of Design Change

- The following plant changes have been incorporated to increase condensate return to the IRWST
  - Add downspouts to polar crane girder and internal stiffener to drain condensate directly to IRWST
    - Minimizes losses associated with re-attaching flow to containment wall and with flow over support plates
  - Optimize IRWST gutter design and location
    - Extended to collect above upper equipment hatch and personnel airlock
  - Changed routing of cables to hydrogen sensors
    - Reduces quantity of support plates (obstacles) attached to the containment dome
- Licensing basis would not have been met without design changes

Safe Shutdown

GDC-34 Requirements

- A residual heat removal (RHR) system must be provided to remove residual heat from the reactor core so that specified acceptable fuel design limits (SAFDLs) and the design conditions of the reactor coolant pressure boundary are not exceeded

- Requires suitable redundancy of the components and features of the RHR system to ensure that the system safety functions can be accomplished, assuming loss of offsite or onsite power, coincident with a single failure.

Safe Shutdown

SECY 94-084 states:

- 420°F is a safe, stable condition for passive plants.
- Other plant conditions constitute a safe, stable state as long as reactor subcriticality, decay heat removal and radioactive materials containment are properly maintained for the long term.
- Passive system capabilities can be demonstrated by appropriate evaluations during detailed design analyses, including
  - A safety analysis to demonstrate that the passive systems can bring the plant to a safe stable condition and maintain this condition and
  - No transients will result in the specified, acceptable fuel design limits and pressure boundary design limit being violated

Safe Shutdown - AP1000 DCD Revision 19

- AP1000 DCD revision 19 has inconsistencies
  - Section 6.3.1.1 "Safety Design Basis" describes PRHR closed loop, "…capability to establish safe shutdown conditions, cooling the reactor coolant system to about 420°F in 36 hours."
  - DCD analysis that demonstrates 420F in 36 hours is not a design basis analysis
- AP1000 DCD revision 19 supporting analyses demonstrate
  - Design meets GDC 34 requirements using Design Basis Analysis (Chapter 15) assumptions
  - Design achieves 420°F in 36 hours using conservative, non-bounding assumptions performance analysis
- Design description revised to establish clear separation of safety design basis from non-safety design features (Performance goal)

Issues Addressed

- Calculation Model Reevaluated
  - Error correction (Spreadsheet vs. LOFTRAN)
  - Simplification
  - Use of LOFTRAN with potential for two-phase flow
    - Heat loss vs. adiabatic analysis assumptions
- Design Basis Accident Extended (DBA) to 72 hours
- Safe Shutdown Analysis Confirmed
  - System capability to cooldown to 420°F in 36 hours
- Long Term PRHR Operation Capability Reevaluated (not indefinite)
- Operational Impacts Assessed

Analysis Conclusions

- Design basis analysis demonstrates:
  - PRHR closed loop cooling can maintain the plant in a safe stable condition for 72 hours
- Conservative, non-bounding analysis demonstrates:
  - PRHR closed loop cooling can cool the RCS to 420°F in less than 36 hours
  - PRHR closed loop cooling can maintain safe shutdown (<420°F) for at least 14 days
  - Adiabatic analysis of the RCS is appropriate

Summary of Licensing Basis Change

DCD Revision 19	Levy FSAR
1. For non-LOCA events, PRHR performance meets all Chapter 15 analysis requirements	1. FSAR Chapter 15 analysis extended to 72 hours
2. Safety design requirement that PRHR cooling can achieve safe shutdown in less than 36 hours.	2. No change in analysis method. FSAR clarifies that this is non-safety design requirement based on conservative, non-bounding analyses
3. PRHR cooling can maintain safe shutdown (SSD) indefinitely.	3. FSAR identifies that PRHR closed loop cooling can maintain SSD for at least 14 days based on conservative, non-bounding analysis

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Other AP1000® Emergent Issues

- Post Accident Main Control Room Operator Dose
- Hydrogen Venting Inside Containment
- Flux Doubling Compliance with IEEE 603
- Main Control Room Heat Up

Post-Accident Main Control Room Dose

Problem Statement:

- The certified design did not include direct dose contributions from the VES filter unit: direct filter dose increase the operator dose when considered
- The Main Steam line break analysis did not model the most limiting release scenario: secondary side coolant release timing assumptions were non-bounding
- Discrepancies were identified in the underlying shielding calculations for post-accident operator dose: AP1000 shielding design non-conservatively differed from the analysis model

Issue Resolution:

- A combination of design and analysis changes were needed to demonstrate operator doses satisfy General Design Criterion (GDC) 19
- Reported doses decrease from DCD Revision 19

Hydrogen Venting Inside Containment

Problem Statement:

- **AP1000** design changes to containment layout were implemented without revision to supporting analyses for hydrogen diffusion flame
- In one particular severe accident scenario (frequency = 6E-9/yr), a hydrogen diffusion flame may create a locally high temperature near containment pressure boundary, hatch and penetrations
  - Analysis required to verify a containment survivability
  - ITAAC revision is required to reflect containment layout design changes

Issue Resolution:

- Updated analysis confirms containment survivability during a hydrogen burn event

Flux Doubling Compliance with IEEE-603

Problem Statement:

- The design did not comply with a portion of IEEE 603 Sub-clause 6.6 criteria:
  - Whenever the applicable permissive conditions are not met, a safety system shall automatically prevent the activation of an operating bypass or initiate the appropriate safety function(s).

Issue Resolution:

- A new permissive, P-8, based on minimum required reactor coolant temperature for criticality (MTC), was added
- Design now complies with IEEE-603

Main Control Room Heat Up

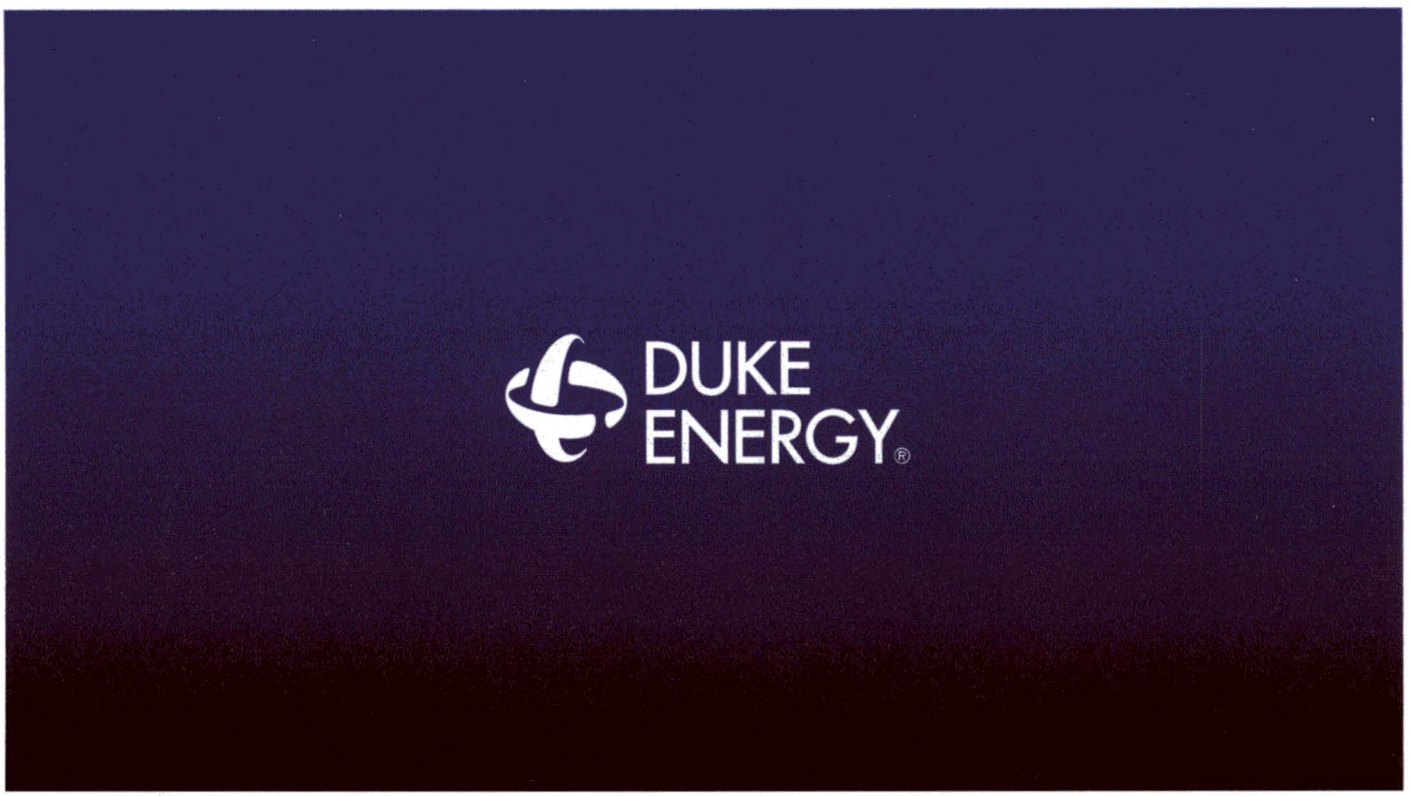
Problem Statement:

- Throughout the design evolution of the MCR, the size and quantity of equipment have increased, raising the total MCR heat load. These increases result in a MCR temperature response exceeding the current licensing basis limit and equipment qualification conditions
- A new more limiting transient where non-safety power is provided to non-safety equipment but VBS is NOT available was identified

Main Control Room Heat Up

Issue Resolution

- Two stage automatic load shed
  - This automatic operation is proposed to maintain the required MCR environmental conditions
    - Only select non-safety loads are de-energized, with no impact to the minimum inventory of displays / controls provided by the primary dedicated safety panel
    - No impact to the plant controls and indication of plant parameters at operator workstations
    - Load shed circuitry is safety related
- Additional Surveillance Requirements
  - Limit initial conditions for adjacent rooms in the updated MCR Heat Up analysis
  - Limit moisture content for air in the VES storage tanks
- Human Factors Considerations
  - Analysis supports unlimited operator stay time at a WBGT Index of 90°F
    - Acceptance criterion is from NUREG-0700
    - Same limit is met for post-72 hour ancillary fan operation
- Evaluation concludes that proposed changes confirm MCR temperature requirements are met and no limits are imposed on personnel stay time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