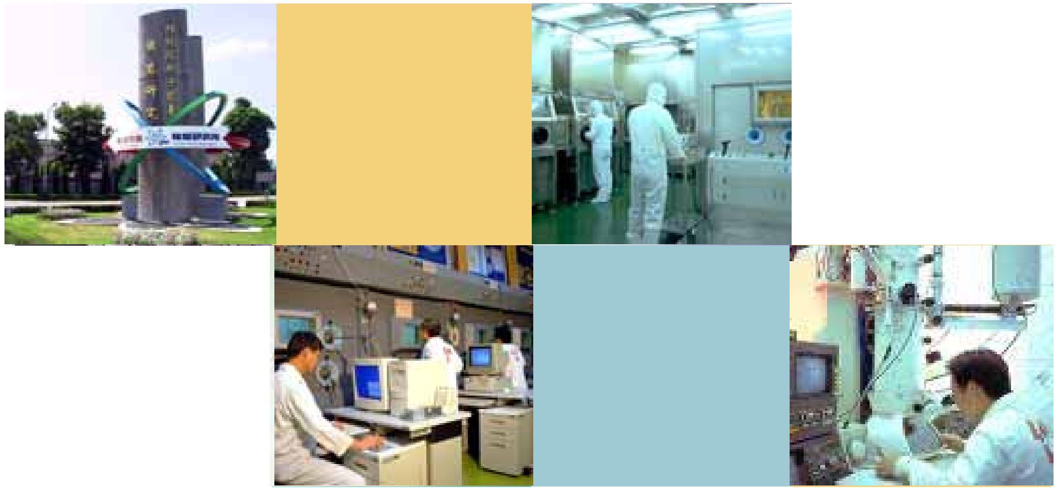
2008 AEC-NRC Bilateral Technical Meeting

Status of Lungmen FSAR Review with Emphasis on Transient and Accident Analysis



May 12-13, 2008

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Institute of Nuclear Energy Research

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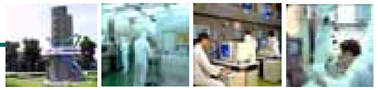
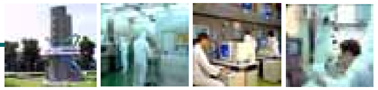


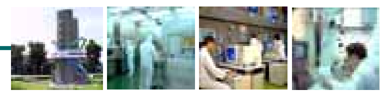
Overview of FSAR Review

1. Review of safety analysis methodology(07/2003~11/2004)
2. Received Taipower's FSAR submittal on August 15, 2007.
3. In response to acceptance review by AEC, Taipower submitted Amendment of FSAR Chapter 7 and 18 in early March, 2008.
4. Review process has been established. Review team consists of AEC staff and members from INER and domestic scholars and experts.
5. Review of three topics (Fire Protection、Seismic Design、Digital I &C) have been contracted to professional organizations or groups



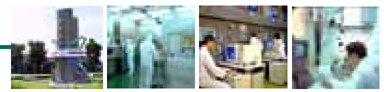
Overview of FSAR Review (cont.)

- Focus of FSAR Review
  - **ABWR Unique Design Features**
  - **Lungmen Unique Design Features**
  - **Limiting Cases**
  - 119 Follow-up Items of PSAR
  - PSAR Amendments
  - Differences Between PSAR and FSAR
  - Differences Between SSAR and FSAR



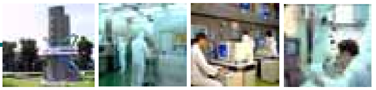
Overview of FSAR Review (cont.)

- Examples of ABWR Unique Design Features
  - **Digital Instrument and Control System**
  - **Reactor Internal Pumps**
  - 3 Independent Safety Systems
  - Fine Motion Control Rod Drive
  - **Compact Containment**
  - Containment Overpressure Protection System
  - AC Independent Water Addition System
  - Lower Drywell Flooder
  - Use of Better Material



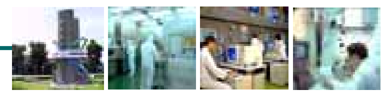
Overview of FSAR Review (cont.)

- Examples of Lungmen Project Unique Design Features
  - Adopts modern fully-integrated digital design for control, communications, and human-system interfaces (HSIs).
  - 110% Rated Steam Flow Bypass Capacity
  - Small Low Population Zone (LPZ)



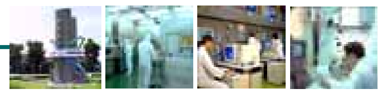
Diversity and Defense-in-Depth Analysis

- Purpose
  - To demonstrate that the design has adequate coping capability in the event of a digital common cause failure
- Types and Consequences of Digital Common Cause Failure
  - Failure to actuate or control without false indications
  - Failure to actuate or control with false indications
  - Partial actuation
  - Spurious actuations



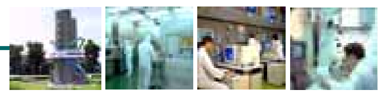
Diversity and Defense-in-Depth Analysis (cont.)

- Branch Technical Position HICB-19
  - For each anticipated operational occurrence in the design basis occurring in conjunction with each single postulated common-mode failure
  - To calculate the plant response using best-estimate (realistic assumptions) analyses
  - No radiation release exceeding 10% of the 10 CFR 100 guideline value
  - No violation of the integrity of the primary coolant pressure boundary



Diversity and Defense-in-Depth Analysis(cont.)

- Concern: To clarify whether or not Lungmen FSAR Ch15 needs to be modified in order to reflect the effects of software common mode failure :
  - GE concludes that BTP-19 does not require that the analysis in Chapter 15 of FSAR to be modified to reflect the effects of software common mode failure. Also, the issue of software common mode failure is considered beyond the design basis from EPRI and industry
  - The requirement that the digital RPS should be protected against CCFs is imposed by the USNRC no matter how the software common mode failure is classified, namely, beyond design basis or not. [ref]



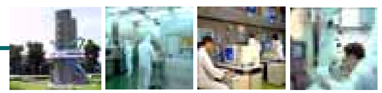
Diversity and Defense-in-Depth Analysis (cont.)

NUREG-1503(Final SER to the certification of ABWR)

- Based upon NUREG-0493, the staff's D3 assessment of the ABWR design was performed by the Lawrence Livermore National Laboratory (LLNL)
- In performing the event-based common-mode failure evaluation, GE evaluated 14 events from the SSAR Chapter 15 transients and accidents in the ABWR I&C system

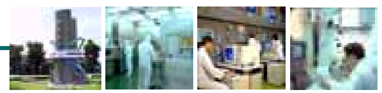
Lungmen specific Review

- The evaluation performed by GE during ABWR certification may not be applicable to the Lungmen power plants.
  - In response to our questions, we were told that GE is preparing a comprehensive evaluation of the control system diversity in the Lungmen plant, including the availability of diverse controls for automated actions, manual actions and the plant response to the accidents and transients described in the Lungmen FSAR.
  - This issue is considered as an open issue in Lungmen specific safety analysis review. Final resolutions need to be reached in the future.

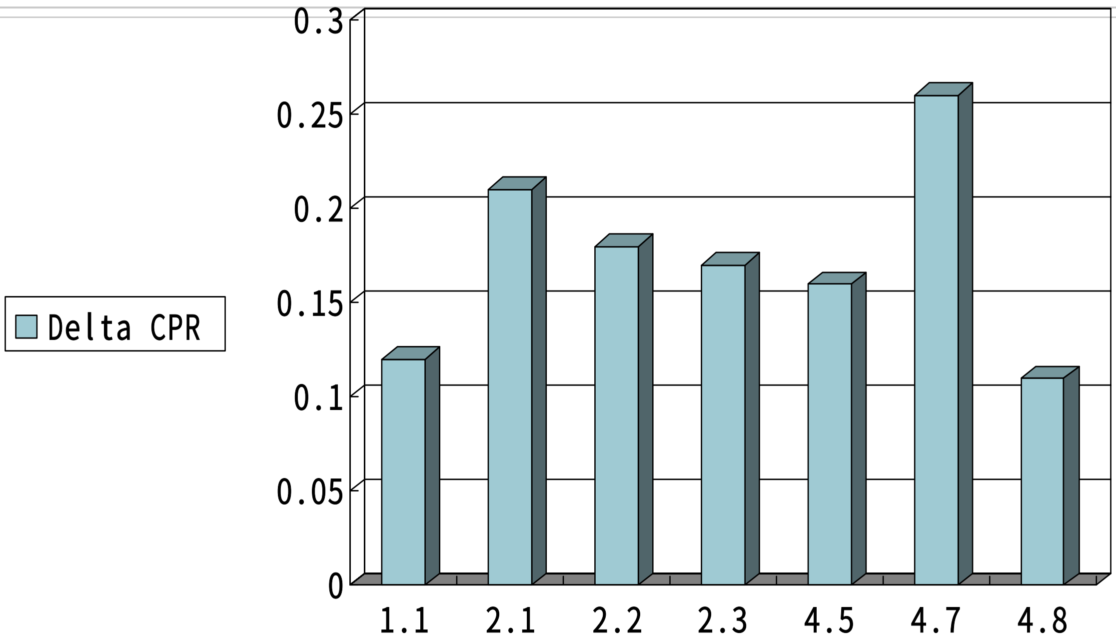


Transient & Accident Analysis Review

1. Decrease in Core Coolant Temperature
2. Increase in Reactor Pressure
3. Decrease in Reactor Coolant System Flow Rate
4. Reactivity and Power Distribution Anomalies
5. Increase in Reactor Coolant Inventory
6. Decrease in Reactor Coolant Inventory



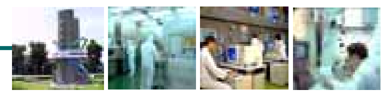
Limiting ΔCPR Events



- 1.1 Loss of Feedwater Heating
- 2.1 **Fast Closure of One Turbine Control Valve**
- 2.2 Load Rejection Without Bypass
- 2.3 Turbine Trip Without Bypass
- 4.5 Fast Runout of All Reactor Internal Pumps
- 4.7 **Mislocated Fuel Bundle Event**
- 4.8 Misoriented Fuel Bundle Event

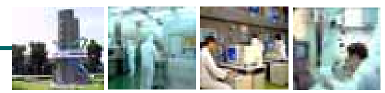
Limiting ΔCPR Events
Increase in Reactor Pressure

- Fast Closure of one TCV
  - The most limiting core-wide transient for Lungmen NPS
  - No direct scram from the fast closure of one TCV
  - Being one of the key parameters, Maximum combined flow limitation should be included in Technical Specification.
  - The need of enhancing the modeling capability of the main steamlines has been identified.
- Load Rejection Without Bypass
  - The scram trip signal initiated by Load Rejection is purposely delayed to allow time for bypass valve operation verification



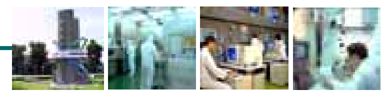
Limiting ΔCPR Events
Reactivity and Power Distribution Anomalies

- Mislocated/Misoriented Fuel Bundle Accident
  - Mislocated Fuel Bundle Event is the most limiting ΔCPR Event for Lungmen NPS
  - More limiting for the initial core
  - The TPC proposed to change the acceptance criteria of these events from ΔCPR to radiological consequences
    - Only EAB radiological consequences have been calculated in the GE's submittal (amendment 28 to GESTAR II). LPZ radiological consequences have not been calculated/evaluated.
  - Specific guidance for this event was not found in the REG GUIDE 1.195 (05/01/2003) "Methods And Assumptions For Evaluating Radiological Consequences Of Design Basis Accidents At Light-Water Nuclear Power Reactors"



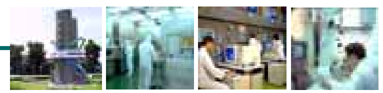
LOCA

- No fuel damage resulted from this accident
  - Peak Cladding Temperature is no longer one of the most important issues
- Radiological Consequences
  - Small LPZ
  - Radiological Consequences of LOCA become one of the most important issues
- Containment Design Limit

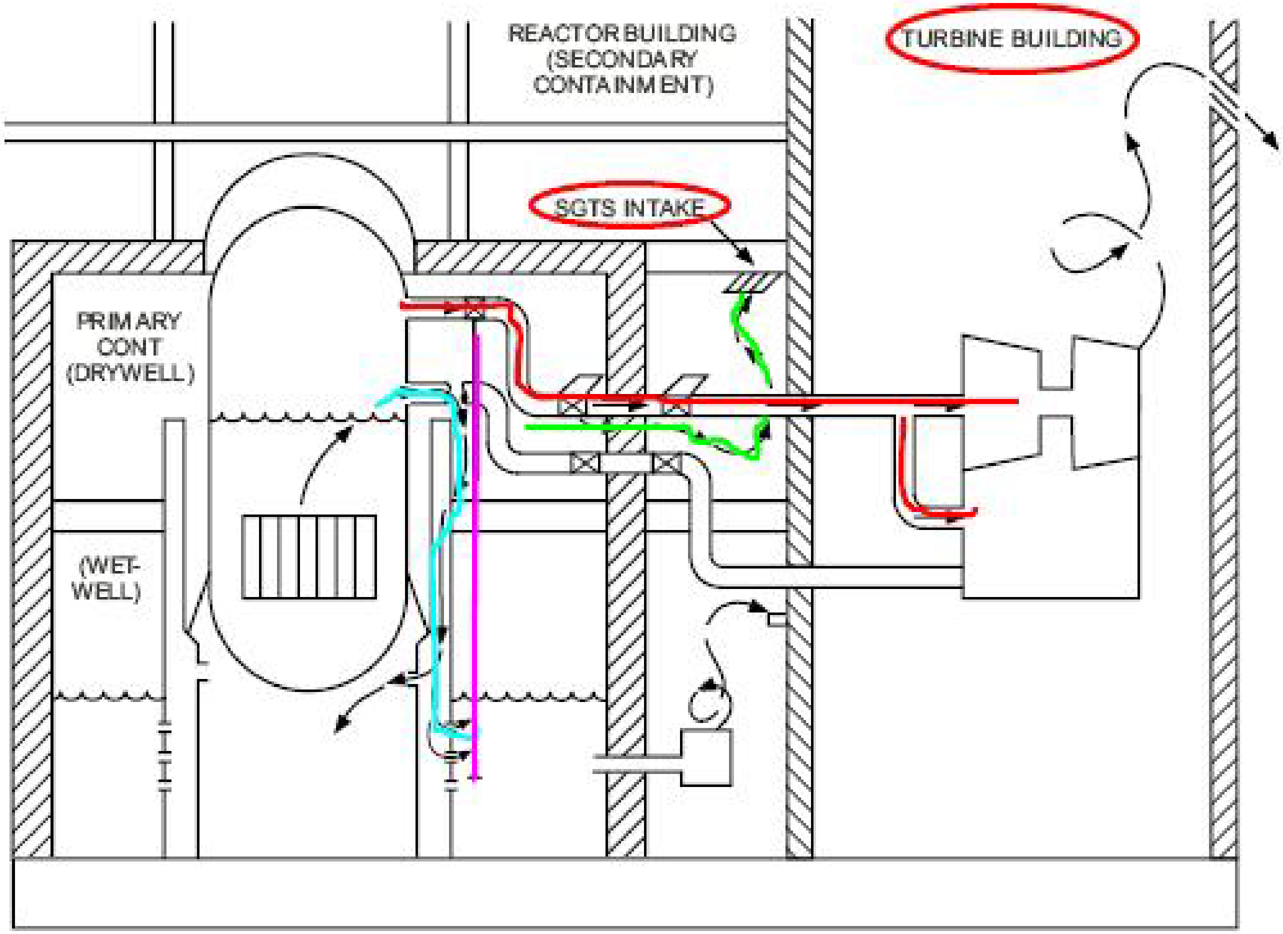


Radiological Consequences of LOCA

- Fission Product Release
  - Based upon the assumptions made in Regulatory Guide 1.3 (Alternate Source Term was not adopted)
  - The fission products found in the core, 100% of the noble gases and 50% of the iodines are released from the core
- Release Pathways
  - **Containment Leakage Contribution**
  - **Main Steam Isolation Valve Leakage**
  - Post-LOCA Leakage Contribution from Engineered Safety Features Systems Outside Containment

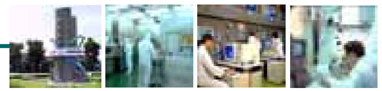


LOCA leakage path



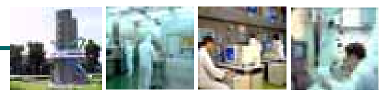
Radiological Consequences of LOCA
LOCA Leakage Path

- Containment Leakage Contribution
  - It is qualified as an elevated release point in accordance with RG. 1.145
  - Flow through the Reactor Building via the Standby Gas Treatment System to the Combined Plant Services Stack (CPSS)
  - CPSS is a seismically qualified safety related stack reaching a height of 141 meters.



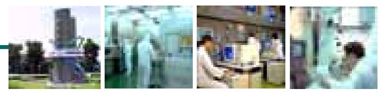
Radiological Consequences of LOCA
LOCA Leakage Path

- Main Steam Isolation Valve Leakage
  - It is a ground release point
  - It is assumed that the most critical MSIV fails in the open position and the other MSIV leaks at the maximum technical specification limit
    - The maximum technical specification limit of MSIV in the Lungmen design is reduced from the standard design (0.72% per day -> 0.12% per day)
  - The main steamlines and drain lines are designed to meet SSE criteria and analyzed to dynamic loading criteria. They are used as mitigative components in the analysis of leakage from the MSIVs.



Containment Design Limit Evaluation During LOCA

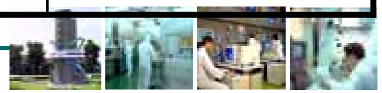
- Drywell airspace temperature exceeds the design limit during the main steam line break event
  - The acceptance criteria of the Standard Review Plan seems to be violated
  - According to the acceptance criteria of SRP section 6.2.1.1.c , “In meeting the requirements of General Design Criteria 16 and 50 regarding the design margin for Mark I, II and III plants at the operating license stage of review, the peak calculated values of pressure and temperature for the drywell and wetwell should not exceed the respective design values”



Containment Design Limit Evaluation During LOCA

- Comparisons of BWR and ABWR containment design
  - Compact ABWR containment

	MARK I	MARK II	MARK III	ABWR
Volume (10^6 ft^3)	0.4	0.5	**1.6**	**0.5**
Thermal capacity (10^9 Btu)	1.7	1.3	1.3	1.3
Design pressure (psig)	62	45	**15**	**45**
LOCA pressure (psig)	44	42	**9**	**39**



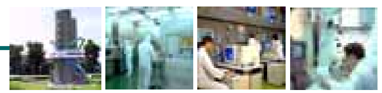
drywell airspace temperature exceeds the design limit

Table 6.2-1 Containment Parameters*

	Design Parameter	Design Value	Calculated Value
1.	Drywell pressure	309.9 kPaG	278.5 kPaG
2.	Drywell temperature	171.1°C	176.3°C
3.	Wetwell pressure	309.9 kPaG	210.4 kPaG
4.	Wetwell temperature		
	• Gas Space	124°C	101.7°C
	• Suppression pool	97.2°C	92.8°C†
5.	Drywell-to-wetwell differential pressure	+172.6 kPaD – 13.7 kPaD	+170.3 kPaD – 3.72 kPaD

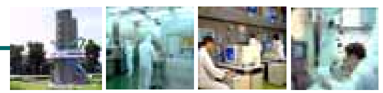
* Although the calculated drywell airspace temperature exceeds the design limit, it does so for only a short time (about 1.1 seconds). Because it takes a much longer time for the drywell structural materials to increase in temperature, the drywell structural materials remain below the design temperature.

† Based on LOCA analysis (FWL and MSL breaks) corresponding to maximum sea water temperature of 35°C and pool initial temperature of 35°C (see Figure 6.2-8).



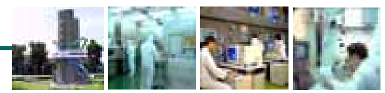
Conclusions and Recommendations

- Focus of FSAR Review has been placed on the unique design features, PSAR follow-up items and limiting cases
- Some issues are considered as open issues in Lungmen specific safety analysis review. Final resolutions need to be reached in the future.
  - Diversity and Defense-in-Depth Analysis and Mislocated Fuel Bundle Event are the two examples.
- We expect new and more specific regulations and guidance to be issued for the above two events. We are very interested in the guidance about how to fulfill the regulation requirements.

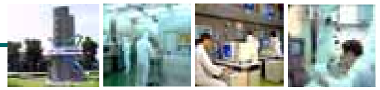


Conclusions and Recommendations

- The needs of enhancing the modeling capability of the main steamlines have been identified for the event of Fast Closure of one TCV and Radiological Consequences of the accidents for all BWRs.
- The need of adding one parameter "Maximum combined flow limitation" to TS has been identified during the analysis of Fast Closure of one TCV
- We expect to have more interactions and strengthened cooperation between both parties.



Thanks for your attention !



Diversity and Defense-in-Depth Analysis
Lungmen Specific Review

- NRC staff's position in NUREG-1503 section 7.2.6
  - The staff considers common-mode software errors to be a special case of single failure and, therefore, protection against such errors is to be part of the design bases.
- NRC staff's position in **DI&C-ISG-02**
  - While the NRC considers common cause failures (CCFs) in digital systems to be beyond design basis, the digital RPS should be protected against CCFs.
- How the software common mode failure is classified shouldn't be the focus of the issue.
- How the requirement can be fulfilled should be the focus of the issue.