RAS 12739

U.S. NUCLEAR REGULATORY COMMISSION
In the Matter of System Energy Resources
Docket No. 52-009-ESP Official Exhibit No. Staff 19
OFFERED by: Applicant/Licensee Intervenor
NRC Staff Other
IDENTIFIED on 1/19/06 Witness/Panel
Action Taken: ADMITTED REJECTED WITHDRAWN
Reporter/Clerk

NRC STAFF EXHIBIT 19

Issue I - Consideration of Radiological Releases

- NRC reviews plant design to ensure shielding and radwaste processing systems are adequate to control doses to public from direct radiation and radioactive effluents within the limits of 10 CFR Parts 20 and 50, Appendix I, and 40 CFR Part 190.
- Releases within these limits do not pose an undue risk to public health and safety.
- ODCM describes methods for control of liquids, gases, and solid waste that may contain radioactive material including radiological effluent and environmental monitoring programs.
- ODCM is reviewed by NRC, and adherence to ODCM is specified in administrative controls section of plant Technical Specifications.

Template=SECY-027

SECY-02

Hearing Issue I
Radiological Reviews and Confirmatory Analyses

- James Wilson
- Eva Eckert Hickey (PNNL)
- Stephen Klementowicz
- Jay Lee
- James V. Ramsdell (PNNL)
- Goutam Bagchi

Issue I - Consideration of Radiological Releases

- Pathways for radiation exposure of the public are evaluated: direct radiation from in-plant and gaseous plume; inhalation; ingestion of water, vegetables, milk, meat, and fish; recreational activities such as swimming.

- ODCM describes the methods for estimating doses to public from these pathways.

- NRC reviews plant design to ensure that occupational radiation exposure can be maintained within the limits of 10 CFR Part 20; Part 20 further requires occupational radiation exposure to be maintained as low as is reasonably achievable.

Issue I - Consideration of Radiological Releases

- NRC reviews the plant design to ensure doses to public can be maintained within the criteria of 10 CFR Part 100, or 10 CFR 50.34 (a)(i) for design basis accidents (loss of integrity of fuel cladding and reactor with intact containment).

- NRC also evaluates probability and consequences of severe accidents (significant core damage and containment failure) to assess overall plant risk.

Issue I - Consideration of Radiological Releases

- The difference between the safety and NEPA environmental reviews results from different regulatory objectives.
- The NEPA reviews are governed by the "rule of reason" and employ a best-estimate methodology to ensure that radiological environmental impact of plant operation will be considered in making licensing decisions.
- The safety review is based on bounding analyses using adverse conditions, resulting in conservative estimates, to ensure that safety design criteria and radiation protection regulations are met.

Overview of Radiological Analysis and Results

Eva Eckert Hickey
Staff Scientist

- Radiological and Uranium Fuel Cycle Impacts
- Technical Expertise:
  - Radiological impacts
  - Non-radiological impacts
  - Uranium fuel cycle

Discussion Topics

- Guidance used for evaluation
- Approach used for review
- Impacts of evaluation using a PPE approach
- Results of evaluation
- Conclusions

Regulatory Standards and Guidance

- Regulations followed
  - 10 CFR Part 51 and NEPA
- RS-002
- NUREG-1555, ESRP
  - 3.5 Radioactive Waste Management System
  - 5.4 Radiological Impacts of Normal Operation
  - 6.2 Radiological Monitoring

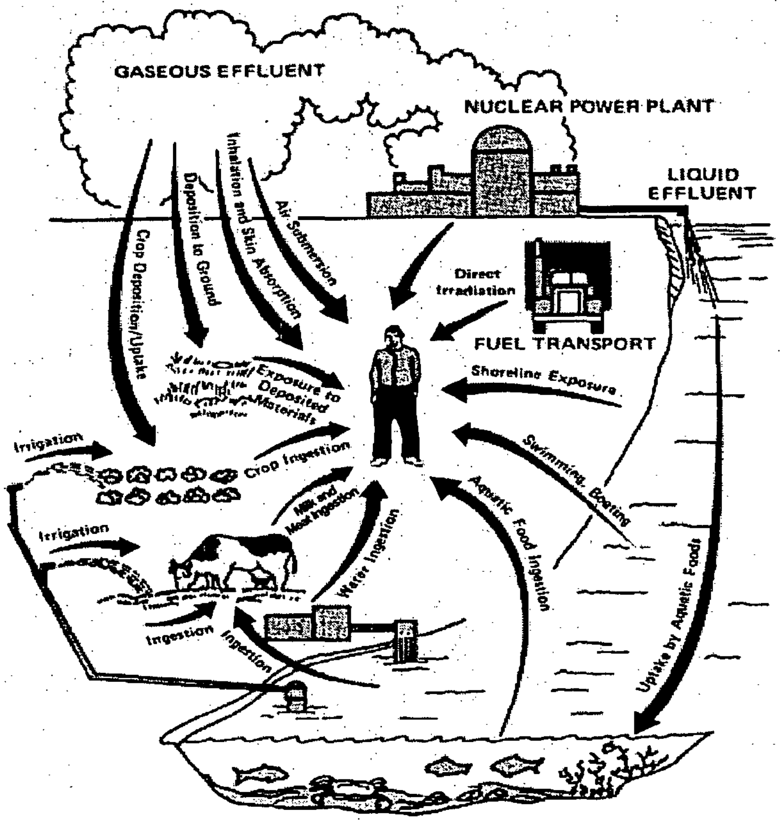
Description of Radiological Environment

- Radiological environmental monitoring program established for GGNS-1
- Pre-operational program (1978-1985)
- Results of annual environmental operating reports
- Annual radioactive effluent release reports
- Doses to maximally exposed individual less than regulatory standards

Radiological Impacts of Operations

- Radiological Health Impacts
  - Public
    - Evaluated dose from gaseous and liquid effluents
    - Performed independent evaluation
    - Doses were within regulatory design objectives and dose standards
  - Workers
    - Occupational dose bounded by current operating LWRs
    - Compliance with 10 CFR 20.1201
    - ALARA
  - Biota
    - Evaluated dose from gaseous and liquid effluents
    - Performed independent evaluation
    - Dose rate estimates were less than NCRP and IAEA studies
  - Staff conclusion – impacts would be SMALL

Exposure Pathways to Humans



Doses from Liquid Effluents

- LADTAP II (NRCDOSE version 2.3.5)
- Source Term
- Reviewed other parameters used
- Estimated total body dose and organ dose
- Compared results to applicant's and to regulatory standards

Liquid Pathway Doses for Maximally Exposed Individual

Pathway	Total Body Dose (adult)	Maximum Organ (bone, child)
Aquatic Foods	2.2 mRem/yr	4.1 mRem/yr
Shoreline Use	0.003 mRem/yr	0.0036 mRem/yr
Total	**2.2 mRem/yr**	**4.1 mRem/yr**

Doses from Gaseous Effluents

- GASPAR II (NRCDOSE version 2.3.5)
- Source Term
- Reviewed other parameters used
- Estimated total body dose and organ dose
- Compared results to applicant's and to regulatory standards

Gaseous Pathway Doses for Maximally Exposed Individual

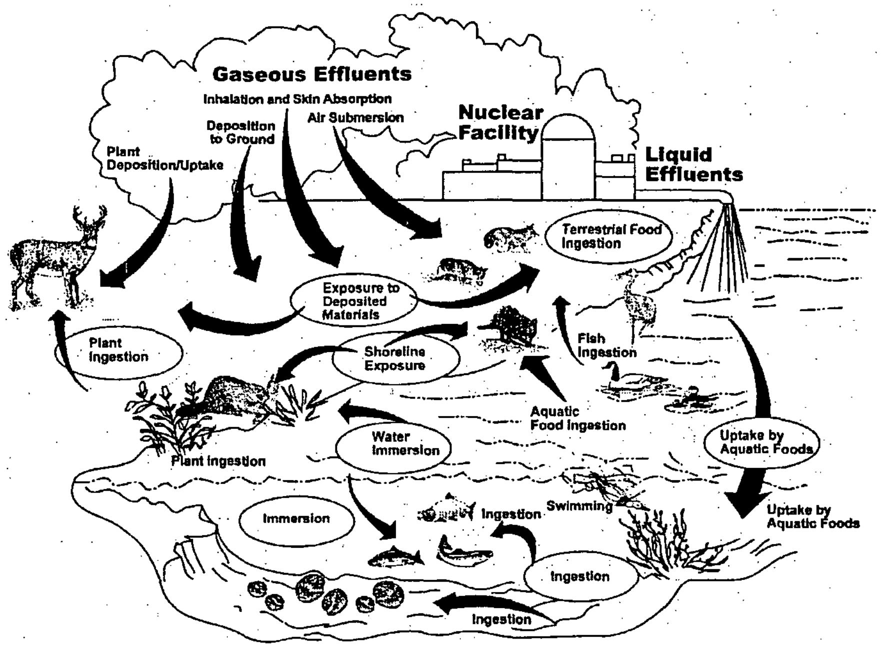
Impacts on Members of the Public

- Whole body and organ dose estimates to MEI from liquid and gaseous effluents are within Design Objectives and 40 CFR Part 190
- Doses at exclusion area boundary from gaseous effluents within Design Objectives
- Thyroid doses are within Design Objectives

Population Dose

- Estimated collective whole body dose within 50 mi ≈3.2 person-rem/yr
- Collective dose from liquid effluent pathway did not include drinking water
- Compared to collective dose from natural background ≈102,000 person-rem/yr

Exposure Pathways to Biota Other than Humans



Doses to Biota

- Liquid pathway for terrestrial and aquatic biota
- Gaseous pathway for terrestrial biota
- LADTAP II and GASPAR II
- Reviewed input parameters
- Compared Staff results to SERI's results
- Compared to regulatory standards

Comparison of Biota Dose to Standards

Biota	Total Dose/unit (mrad/yr)	40 CFR190 (mRem/yr)	IAEA/ NCRP (mrad/yr)
Fish	25	25	365,000
Invertebrate	165	25	365,000
Algae	148	25	365,000
Muskrat	83	25	36,500
Raccoon	21	25	36,500
Heron	195	25	36,500
Duck	83	25	36,500

Radiological Impacts

- Exposures to the public and to workers
  - Estimated doses to public well within regulatory design objectives and standards
  - No observable health impacts to public
  - Occupational doses estimated to be slightly lower than those from current reactors
- Impacts to biota evaluated and found to be acceptable
- Conclusion – radiological impacts from construction and operation would be SMALL

Radiological Safety Evaluation

- Staff used the calculated radiological dose values contained in the Environmental Impact Statement (NUREG-1817) as the basis for its Safety Evaluation Report
- Basis: The Radiological Safety Evaluation and the Environmental Impact Statement overlap on the radiological dose acceptance criteria contained in NRC regulations and guidance.

Grand Gulf Early Site Permit
ASLB Hearing
Hearing Issue I

Jay Lee

Senior Health Physicist

Office of Nuclear Reactor Regulation

Hearing Issue I
Item 1

Selection of the Design Basis Accidents and Event Names That Appear in the SSAR, FSER, and FEIS.

The Staff used the design basis accident (DBA) names that are listed and analyzed in:

(1) RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors (July 2000),"

(2) Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms (July 2000),"

(3) NUREG-0800, "Standard Review Plan for Review of Safety Analysis Report for Nuclear Power Plants (July 1981)," and

(4) NUREG-1555, "Standard Review Plan for Environmental Review for Nuclear Power Plants (October 1999)."

Design Basis Accident Names

Safety Evaluation Report (SER)	Final Safety Evaluation Report (FSER)	Final Environmental Impact Statement (FEIS)
Main Steam Line Breaks (PWR/BWR)	PWR Main Steam Line Break BWR Main Steam Line Break	Main Steam Line Break
Reactor Coolant Pump Locked Rotor (PWR)	Locked Rotor Accident (Reactor Coolant Pump Shaft Break) (note 1)	Reactor Coolant Pump Rotor Seizure (Locked Rotor)
Control Rod Ejection (PWR)	PWR Rod Ejection Accident	Rod Ejection
Control Rod Drop (BWR) (note 2)	BWR Control Rod Drop Accident (note 2)	(note 2)
Small Line Break Outside Containment (PWR/BWR)	Failure of Small Lines Carrying Primary Coolant Outside Containment (note 3)	Failure of Small Lines Carrying Primary Coolant Outside Containment
Steam Generator Tube Rupture - SGTR (PWR)	PWR Steam Generator Tube Failure	Steam Generator Rupture
Loss of Coolant Accident (PWR/BWR)	PWR and BWR Loss of Coolant Accidents	Loss of Coolant Accident
Fuel Handling Accident (PWR/BWR)	Fuel Handling Accident	Fuel Handling (Accident)

Note 1

- The reactor coolant pump shaft break and locked rotor accident assume the same accident sequence after the initiation and result in the same radiological consequence.
- NUREG-0800 listed this event as "Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break" which postulates as an event an instantaneous seizure of the rotor or break of the shaft of a reactor coolant pump (see NUREG-0800 Section 15.3.3 -15.3.4).
- ASLB Inquiry Nos. 81 and 82, "Why is the Reactor Coolant Pump Shaft Break excluded from the staff's review?" It is listed as the Reactor Coolant Pump Locked Rotor Accident.

Note 2

- The SSAR and FSER both listed the BWR Control Rod Drop Accident for completeness but neither the Applicant nor the Staff analyzed the radiological consequence for this event since the certified ABWR includes several unique features that preclude the occurrence of this event (see NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design).

- The ABWR design was certified without the radiological consequence analysis for the Control Rod Drop Accident.

Note 3

- NUREG-0800, NUREG-1555, the FEIS, and the FSER include a DBA titled "Failure of Small Lines Carrying Primary Coolant Outside Containment," while the SSAR listed the same DBA as "Small Line Break Outside Containment."

Hearing Issue I
Item 2

An Overview of the Radiological Analyses

The Applicant

- did not select a particular reactor design
- used surrogate reactor designs (ABWR and AP1000)
- did not perform a new radiological consequence analysis
- directly extracted the radiological consequence analysis results from design certification documentations previously submitted to and reviewed by the NRC in connection with the design certification applications
- provided one DBA (LOCA) for ACR-700 (bounded by AP1000 LOCA)

The Staff

- Performed independent confirmatory review at the time of the design certifications (ABWR in 1994 and AP1000 in 2004)

- Did not need to perform further confirmatory radiological consequence analyses in review of the Grand Gulf ESP application (FSER Section 15.3.4)

- Verified the Applicant's calculations using Case 1 and Case 2 equations

Radiological Consequence Dose Calculation for Design Basis Accidents

➢ Standard Reactor Certification (Internal Exposure as an example)

Radiation Dose dc (rem) = Source Term dc (Ci) x Atmospheric Dispersion Factor dc (sec/m3) x Breathing Rate (m3/sec) x Dose Conversion Factor (rem/Ci) ....(Equation 1)

- **Radiation Dose dc : Radiation dose in standard reactor certification document .**
- **Source Term dc : Source term in standard reactor certification document.**
- **Atmospheric Dispersion Factor dc : Postulated atmospheric dispersion factor in standard reactor certification document .**
- **Breathing Rate : Breathing rate for standard man in International Commission on Radiation Protection, Publication II (1959).**
- **Dose Conversion Factor : Dose conversion factor in Federal Guidance Report 11, "Limiting Values of Radionuclide Intake And Air Concentration and Dose Conversion Factors for Inhalation, Submersion, And Ingestion (1988)," U.S. Environmental Protection Agency and Oak Ridge National Laboratory.**

Grand Gulf ESP
Site Radiation Dose Estimation
(Internal Exposure)

- **Radiation Dose esp (rem) = Source Term dc (Ci) x Atmospheric Dispersion Factor esp (sec/m3) x Breathing Rate (m3/sec) x Dose Conversion Factor (rem/Ci) .... (Equation 2)**

Radiation Dose esp : Radiation dose at the proposed ESP site.

Source Term dc : The source term from the standard reactor certification document.

Atmospheric Dispersion Factor esp : Site-specific atmospheric dispersion factor at the proposed ESP site (these values are Site Characteristics).

Applicant's Method (Case 1)

- Substituting Equation 1 into Equation 2 for Source Term dc (Ci)

- Radiation Dose esp (rem) = Radiation Dose dc (rem) x Atmospheric Dispersion Factor esp (sec/m3) x Breathing Rate (m3/sec) x Dose Conversion Factor (rem/Ci) / [Atmospheric Dispersion Factor dc (sec/m3) x Breathing Rate (m3/sec) x Dose Conversion Factor (rem/Ci)]

Equations Used by the Applicant

CASE 1

- Radiation Dose esp (rem) = Radiation Dose dc (rem) x Atmospheric Dispersion Factor esp (sec/m3) / Atmospheric Dispersion Factor dc (sec/m3)

CASE 2

- Radiation Dose esp (rem) = Source Term dc (Ci) x Atmospheric Dispersion Factor esp (sec/m3) x Breathing Rate (m3/sec) x Dose Conversion Factor (rem/Ci)

Conclusion

- The Staff concludes that the Applicant demonstrated the suitability of the proposed ESP site, by meeting the dose consequence evaluation factors set forth in 10 CFR 50.34 (a)(1) and 10 CFR 100.21.

- Therefore, the Staff further concluded that the Applicant complied with the requirements in 10 CFR Part 52.17.

Overview of Radiological Analysis and Results

James V. Ramsdell, Jr
Staff Scientist

- Impacts of Postulated Accidents

- Technical Expertise:
  - Meteorology/Climatology
  - Atmospheric Dispersion
  - Consequence Assessment

Discussion Topics

- Design Basis Accident Review
  - Process
  - Results
- Severe Accident Evaluation
  - Process
  - Results
  - Externally-Initiated Events
- Conclusions

Regulatory Standards and Guidance

- Regulations
  - 10 CFR Parts 50, 51, 52, and 100
- Guidance
  - Standard Review Plans (RS-002, NUREG-0800, NUREG-1555)
  - Regulatory Guides 1.3, 1.23, 1.145, 1.183, 4.2
- Other
  - NUREG-1437 (License Renewal GEIS)

Design Basis Accidents

- Reactor Input (ABWR, AP1000, ACR-700)
  - Accident Selection
  - Source Terms (Isotope by time)
  - Design Dispersion Factors
  - Design Doses
- Site-Specific Input
  - Adverse and Typical Atmospheric Dispersion Factors
  - EAB and LPZ boundaries
- Dose Criteria (10 CFR Parts 50 and 100, NUREG-0800)

Design Basis Accident TEDE Dose Estimates

- FSER (Adverse Meteorological Conditions)
- FEIS (Typical Meteorological Conditions)

	Review Criteria	FSER (Sv)		FEIS (Sv)	
	(Sv)	EAB	LPZ	EAB	LPZ
ABWR LOCA	2.5×10^{-1}	4.3×10^{-2}	1.0×10^{-1}	5.9×10^{-3}	5.4×10^{-2}
ABWR MSLB Pre-existing I_2 spike	2.5×10^{-1}	1.1×10^{-2}	1.6×10^{-3}	1.4×10^{-3}	4.8×10^{-4}
AP1000 LOCA	2.5×10^{-1}	2.46×10^{-1}	6.5×10^{-2}	3.4×10^{-2}	2.2×10^{-2}
AP1000 SGTR Pre-existing I_2 spike	2.5×10^{-1}	3.0×10^{-2}	2.3×10^{-3}	4.1×10^{-3}	7.5×10^{-4}

Severe Accident Evaluation

- Reactor Input (ABWR, AP1000)
  - Release Categories and Core Damage Frequencies
  - Source Terms
- Site-Specific Input
  - Meteorological Data
  - Land-Use Data
  - Population Data
- Evaluation Criteria (Risk)
  - Commission's Safety Goals
  - Comparison With Current Generation Reactors

MACCS2 Computer Code

- NRC/DOE Standard Code
- Isotopic Source Term (60 Isotopes)
- Site-Specific Land Use and Population Data
- Hourly Site-Specific Meteorological Data
- Time-Dependent Dispersion and Deposition Model
- Simple Evacuation Model
- Probability Estimates of Doses, Health Effects and Economic Impacts

Severe Accident Analysis

- Core Damage Frequency From Internally-Initiated Events (Design Certification)
- Dose, Health, or Economic Consequence (MACCS2)
- Probability Weighted Consequence—RISK (CDF x Consequence)

Severe Accident Risks

- ABWR
- AP1000
- Commission Safety Goals

Severe Accident Risks

- ABWR
- AP1000
- GGNS
- Median Current Generation Nuclear Power Plant

External Initiating Events

- ABWR and AP1000
  - Considered but Numerical CDFs not Adopted
  - Characterized as "Extremely Small"
- Current Generation Reactors
  - NUREG-1742, *Perspectives Gained From the Individual Plant Examination of External Events (IPEEE) Program*
  - CDFs Typically Same Magnitude as or Smaller Than for Internal Initiating Events

Cumulative Impacts

- Design Basis Accidents
  - Impacts Based on Individual Reactors
  - No Simultaneous Accidents
  - Cumulative Impacts not Evaluated
- Severe Accidents
  - Risk of Severe Accident at GGNS
  - Risks of ABWR and AP1000 Reactors
  - Cumulative Risk ≈ GGNS Risk

Conclusions

- Potential Impacts of Design Basis Accidents for Advanced Light Water Reactors are Within Regulatory Criteria Designed to Protect Public Health and Safety. Therefore, the impacts are of SMALL significance.
- Severe Accident Risks for Advanced Light Water Reactors are Within the Commission's Safety Goals. Therefore, the impacts are of SMALL significance.

Accidental Release Issues

- Permit Condition 2 is technically feasible
  - Location and design
- Ground water monitoring is not required for accidental release:
  - Release point and source are identifiable with the plant
  - At the ESP stage, reliable radionuclide transport characteristics can not be established for an effective monitoring plan

Goutam Bagchi

Accidental Release Issues

- Radwaste tank failure analysis for COL is not necessary, regulatory guidance exist.
  - Seismic Category I design criteria
  - Radwaste inventory is located on nuclear island
  - Reinforced concrete storage surfaces are sealed against liquid radwaste seepage
  - Design to incorporate spillage containment

Goutam Bagchi