GE Hitachi Nuclear Energy

EVOLUTION OF “UNLIKELY” ASSESSMENT IN NCS ACCIDENT SEQUENCES

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Track 1: Development of Standards and Assessment Methodology

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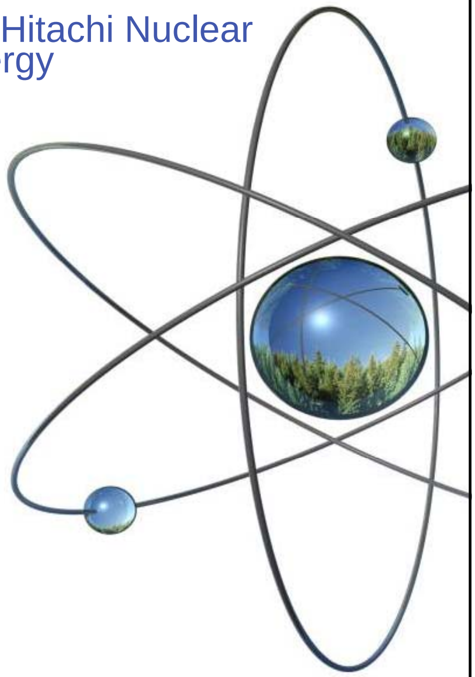

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Overview

- **Introduction to GE Hitachi Nuclear Energy (GEH)**
- **GEH Wilmington Site & Regulatory Framework**
- **Unlikely NCS Risk Assessment: Historic Expert Era**
  - **ANSI/ANS-8.1 Fundamentals**
  - **Qualitative Assessment Process**
- **Highly Unlikely NCS Risk Assessment: Modern QRA Era**
  - **Quantitative Risk Assessment (QRA) Methodology**
  - **Unmitigated Frequency**
  - **IROFS Failure Probability**
  - **Quantify Overall Likelihood, L_T**
  - **Practical Example**
- **Conclusions**

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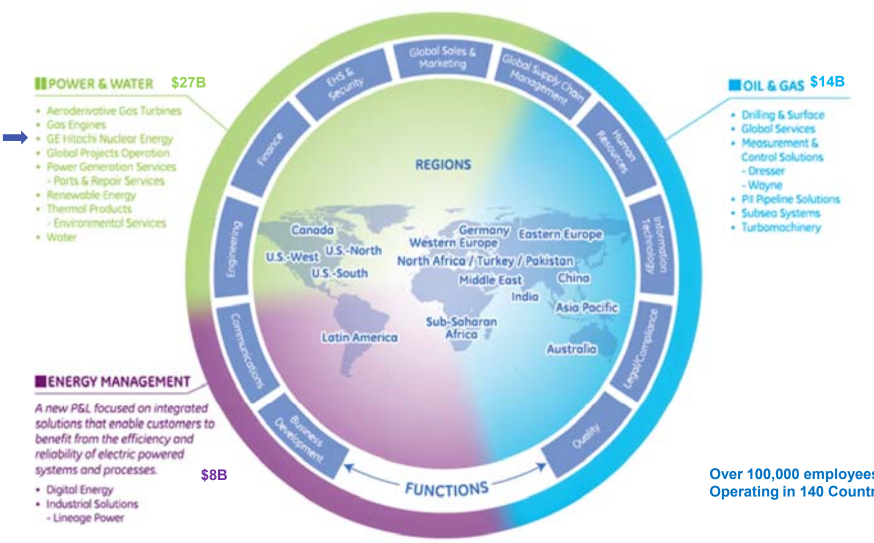
Introduction to GEH

GE Energy



Over 100,000 employees
Operating in 140 Countries

GE Hitachi Nuclear Energy

Nuclear Power Plants

- Generation III Advanced Boiling Water Reactor
- Generation III+
- Simulator & Executive training and consulting

Nuclear Services

- Reactors, turbines & balance of plant
- Life extension
- Power uprates
- Performance services
- Outages and inspections

Fuel Cycle

- Boiling water reactor & mixed oxide fuel
- GE Hitachi Canada Candu fuel & handling equipment
- Fuel engineering services
- Enrichment
- Nuclear isotopes

GEH Nuclear Fuel Cycle

GLE scope
Parent scope
Out of scope

Fuel fabrication
- BWR Fresh Fuel
- BWR RU Fuel
- CANDU Fuel
- MOX Fuel

Re-conversion to UO_2 powder

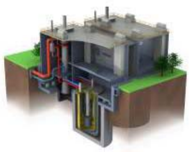
Nuclear plant operation

Core design
Fuel design
Mixed oxide fuel design

Spent fuel storage

UF_6 enrichment

Spent fuel reprocessing
(Advanced technologies)



Conversion to UF_6 gas

Uranium mining

GEH Licensed Nuclear Activities

Wilmington

UF6 Conversion

UO2 Powder and BWR Fuel Fabrication

Nuclear Packaging

GLE Classified Technology

Kurihama

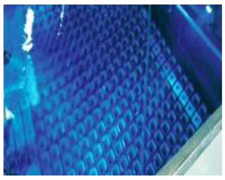
Nuclear Material and Fuel Fabrication

Vallecitos

Spent Fuel Storage

Operating Nuclear Test Reactor

Hot Cell / Lab Facilities



Picture does not depict fuel stored at Morris

Morris

Spent Fuel Storage Facility

Canada

Natural Uranium Processing and Fuel

Fabrication / LEU License 2010

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Wilmington, NC

Site History

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Wilmington Site History

- 1,650 acres (300 developed)
- Over 2 million manuf. square feet
- Over 2,000 employees

Ground breaking	GE Aircraft Engines	Fuel collocation	Dry powder conversion	GNF JV formation	GENE HQ relocation	GEH JV formation	GEH, Cameco venture
1967	1980	1994	1997	2000	2003	2007	2008

10CFR70 ISA rule published

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GEH Regulatory Regime

The GE Hitachi Nuclear Energy's (GEH) Global Nuclear Fuel – America's (GNFA) BWR fuel manufacturing and Global Laser Enrichment (GLE) experimental Test Loop process operations are governed by USNRC special nuclear material license SNM-1097.

Nuclear criticality safety program commitments contained in the license are premised on applicable 10CFR70 regulations, ANSI/ANS-8 series national consensus standards, and Regulatory Guide 3.71 exceptions to these standards.

Nuclear Business Regulation

Nuclear criticality **risk assessment** is a continuous process that must be balanced against other EHS safety & security disciplines

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Unlikely Assessment:
Historic Era
[1960s – late 1990s]

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ANSI/ANS-8.1

4.1.2 Process Analysis.

Before a new operation with fissionable material is begun, or before an existing operation is changed, it shall be determined that the entire process will be subcritical under both normal and credible abnormal conditions

Fundamental U.S. treatise #1

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ANSI/ANS-8.1

4.2.2 Double Contingency Principle.

Process designs should incorporate sufficient factors of safety to require at least two <u>unlikely</u>, independent, and concurrent changes in process conditions before a criticality accident is possible."

Fundamental U.S. treatise #2

Unlikely NCS Risk Assessment

CSA

- Criticality Safety Analysis (CSA) written to establish subcritical limits on controlled parameter(s)
- Demonstrate Double Contingency Principle license condition is satisfied

NRC Review

- NRC staff sit down with responsible CSE and discuss documented CSA via "page turn" exercise
- Develop facility "technical practices" knowledge

Field Review of Controls

- NRC staff perform "as-built" walk down of process & review implemented controls in facility with CSE
- Evaluate as-built condition vs. CSA assumptions; independently evaluate if DCP met in field
- Qualitative but robust NCS risk assessment [no NCS accident in U.S. since ~1978]

A simple, elegant approach to NCS risk

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Historical Evaluation of “Unlikely”

This historical approach was considered “expert based” and relied heavily on practical facility-specific NCS experience.

These were the proverbial “good old days” since the demonstration of double contingency was considered sufficient and adequate to render the postulated criticality accident sequence risk “unlikely” in a qualitative, but rigorous fashion.

In addition to above, the new 10CFR70 rule, effective OCT2000, fundamentally changed the above historical approach used in the U.S.

Highly Unlikely Assessment: **Modern Era** [2000s – present]

Quantitative Risk Assessment (QRA) Era

Nuclear criticality accident sequences are deemed HIGH CONSEQUENCE events (S=3).

Item Relied On For Safety (IROFS) mean structures, systems, equipment, components, and activities of personnel that are relied on to prevent potential accidents at a facility that could exceed the performance requirements in § 70.61 or to mitigate their potential consequences… [10CFR

Severity of Consequences	Likelihood of Occurrence: Likelihood Category 1 Highly Unlikely (1)	Likelihood Category 2 Unlikely (2)	Likelihood Category 3 Not Unlikely (3)
Consequence Category 3 – High (3)	Acceptable Risk 3	Unacceptable Risk 6	Unacceptable Risk 9
Consequence Category 2 – Intermediate (2)	Acceptable Risk 2	Acceptable Risk 4	Unacceptable Risk 6
Consequence Category 1 – Low (1)	Acceptable Risk 1	Acceptable Risk 2	Acceptable Risk 3

For each accident sequence having an unmitigated risk of unacceptable, IROFS must be assigned and the overall mitigated likelihood determined for each accident sequence.

Quantitative Risk Assessment (QRA) Era

Modern determination of "highly unlikely" for credible NCS accident sequences has itself matured:

- Original NRC-approved OL methodology – simple form (2002):

$$L_T = \sum_{i=1}^{i=N-1} (\lambda_{f,i} + \lambda_{d,i}) \bullet \lambda_{cmf,i}] + [\lambda_{f,N}]$$

- After ~5 years of implementation, regulator expectations changed….
- Final GEH NRC-approved OL methodology - general form (2007-current):

$$L_T = \lambda_{IE} + \sum_{k=1}^{M} P_{E,k} + [\sum_{i=1}^{i=N-1} (\lambda_{f,i} + \lambda_{(T/2+MTTR),i}) \bullet \lambda_{IND,i}] + [\lambda_{f,N} + \lambda_{(T/2+MTTR),N}]$$

or equivalently, $L_T = F_{IE} \prod_{j=1}^{J} F_{CE,j} \prod_{i=1}^{I} F_{EC,i} \prod_{k=1}^{K} P_{IROFS,k}$

This quantitative approach developed by GEH is rigorous and adequately covers p-type and f-type NCS controls

QRA Era: Unmitigated Frequency

The unmitigated NCS accident sequence frequency is determined by multiplying the initiating event frequency by the conditional event and enabling condition probabilities:

$$F_{UL} = F_{IE} \prod_{j=1}^{J} F_{CE,j} \prod_{i=1}^{I} F_{EC,i}$$

Where

F_{UL} = unmitigated likelihood (consequence frequency)

F_{IE} = initiating event frequency

$P_{CE,j}$ = probability of jth conditional event

$P_{EC,i}$ = probability of ith enabling condition

	Likelihood Category	Frequency of Occurrence
Not Unlikely*	3	More than or equal to 10^{-3} per-event per-year
Unlikely	2	Between 10^{-3} and 10^{-4} per-event per-year
Highly Unlikely	1	Less than or equal to 10^{-4} per-event per-year

* Default selection in absence of quantitative assessment.

For existing GNFA facility, if F_{UL} is ≤ 1.0e-04/year, no NCS IROFS needed

For planned new GLE CF, if F_{UL} is ≤ 1.0e-05/year, no NCS IROFS needed

QRA Era: Probability of IROFS Failure

The probability of failure for an engineered IROFS can also be calculated using the following formulas:

$$P_{IROFS} = \sum_{x=1}^{X} (R_{C,x} + T/2) + \sum_{y=1}^{Y} PFOD_{C,y}$$

Where,

P_{IROFS} = Probability of failure of engineered system (IROFS)

$R_{C,x}$ = Failure rate of xth component of the engineered system

$PFOD_y$ = Probability of failure on demand of the yth component of the engineered system

T_k = Inspection frequency (for passive) or test frequency (for active) of the engineered system

Numerous references exist for IROFS failure probabilities; Human Reliability Analysis (HRA) tools can be used to quantify the human failure events associated with any IROFS that is an administrative control.

QRA Era: Quantify NCS Accident Sequence

The overall likelihood, L_T, of each NCS accident sequence with IROFS applied to prevent or mitigate the consequence can be calculated by using an event tree or the following:

$$L_T = F_{IE} \prod_{j=1}^{J} F_{CE,j} \prod_{i=1}^{I} F_{EC,i} \prod_{k=1}^{K} P_{IROFS,k}$$

An event tree <u>should be provided</u> that illustrates the accident sequences and the pathways to safe end states as well as accident end states.

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Practical Example

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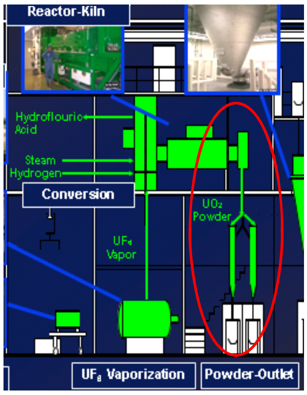
QRA Example: Powder Outlet

Consider an accident sequence in which high moisture UO2 powder is discharged from the DCP reactor-kiln into unfavorable geometry cooling hoppers and/or discharge unicone container.



QRA Example: Powder Outlet

There is no documented occurrence of either event, high moisture powder in the cooling hopper or discharged to a container, during the ten-year operating history of the current hatch valve configuration. In order to avoid using zero occurrences per year, a value of ½ an occurrence in ten years was used. The initiating event frequency is therefore:

F_{UL} = $F_{IE,steam}$ = 0.5 occurrence / 10 year = 5.0 x 10-2/year

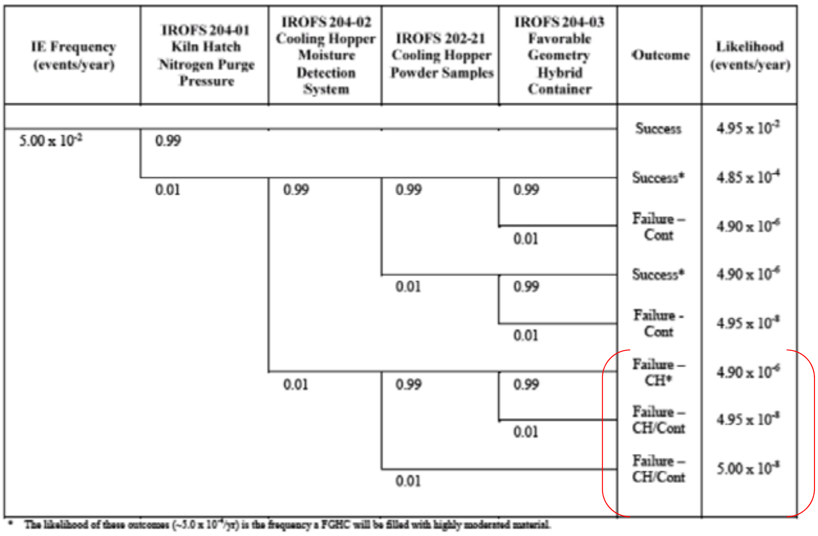
Since F_{UL} is > 1.0e-04/year, <u>IROFS are required</u> to mitigate this credible GNFA NCS accident sequence.

QRA Example: Powder Outlet

The Items Relied on For Safety (IROFS) for this accident sequence includes the following [assigned IROFS failure probability shown, per year basis]:

- **IROFS01**, Kiln Hatch Nitrogen Purge Pressure -The safety function of this augmented administrative control IROFS is to prevent uncontrolled moderator introduction into the cooling hopper by sequence of valve operations supported by operator shutdown of the conversion process upon indication of continued inability to maintain pressure between the hatch valves [1.0 x 10-02].
- **IROFS-02**, Cooling Hopper Moisture Detection System - The safety function of this active engineered control IROFS is to shut the double block valves on the steam feed to the reactor and kiln, direct the steam to the DCP roof vent, thus preventing uncontrolled moderator introduction into the cooling hopper [1.0 x 10-02].
- **IROFS03**, Cooling Hopper Powder Samples - The safety function of this administrative control IROFS is to ensure that high-moisture material is not discharged to a unicone [1.0 x 10-02].
- **IROFS04**, Favorable Geometry Hybrid Container (FGHC) - The safety function of this administrative control IROFS is to provide a discharge path for potentially high moderated material to a favorable geometry container for criticality prevention [1.0 x 10-02].

QRA Example: Event Tree (CH focus)



* The likelihood of these outcomes (~5.0 x 10^{-4}/yr) is the frequency a FGHC will be filled with highly moderated material.

QRA Example: Powder Outlet (CH focus)

Results:

The event tree has two types of failed states, loss of moisture control in the cooling hopper (CH) and loss of moisture control in the discharge container. The QRA result for the CH (only) is summarized below:

$$L_T = F_{IE} \prod_{j=1}^{J} F_{CE,j} \prod_{i=1}^{I} F_{EC,i} \prod_{k=1}^{K} P_{IROFS,k}$$

L_{T1} (Cooling Hoppers) = 0.5 *0.01*0.01*0.99*0.99 $= 4.90 \times 10^{-6}$

L_{T2} (Cooling Hoppers) = 0.5 *0.01*0.01*0.99*0.01 $= 4.95 \times 10^{-8}$

L_{T3} (Cooling Hoppers) = 0.5 *0.01*0.01*0.01 $= 5.00 \times 10^{-8}$

L_T (Cooling Hoppers) $= 4.90 \times 10^{-6}$

Conclusions

Conclusions

- **This paper has contrasted the historical *evolution* of "unlikely" NCS risk assessment methods at GEH from the former double contingency approach to the modern quantitative approach.**
- **The historic "expert era" approach provides a simple, elegant defense of double contingency principle; it is however qualitative in nature, and therefore is highly dependent on the experience level of criticality safety practitioners and regulators alike.**
- **The more modern "quantitative" era provides a more systematic approach and requires the criticality safety practitioner to accurately describe postulated [credible] NCS accident sequences and quantify the true risk of criticality using PRA techniques.**
- **Though compliance with 10CFR70.61 performance requirements can be qualitatively met; GEH has chosen the quantitative risk assessment (QRA) path to demonstrate high consequence [criticality] accident sequences are "highly unlikely".**
- **Future ANSI/ANS-8 series national consensus standards development in the U.S. should consider development of quantitative NCS risk assessment guidance to help unify approaches used at non-reactor facilities. This paper has presented one such approach that is consistent with NUREG-1520.**