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Revision 2

Light Water Reactor Sustainability Program

Report on the Creation and Progress of the Hydrogen Regulatory Research Review Group

August 2023

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Office of Nuclear Energy

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SUMMARY

The current U.S. nuclear generation fleet is increasingly recognized by governmental, scientific, public policy, and industrial communities as having a strategic role in support of the ongoing national transition to a clean energy future.

Federal incentives and actions are aligning to expand the role of nuclear power as a viable and more flexible contributor to the evolving national clean energy mix through programs and initiatives such as nuclear power loan guarantees; the Inflation Reduction Act's (IRA) clean nuclear electrical, steam, and hydrogen incentives; the Infrastructure Investment and Jobs Act (IIJA, also referred to as the Bipartisan Infrastructure Act or BIL); and near-term Department of Energy (DOE) funding opportunities related to nuclear-based hydrogen hubs and nuclear integrated hydrogen demonstration projects.. Additionally, leveraging clean nuclear electricity and steam is being explored by industries desiring to transition away from carbon-intensive energy sources.

Even with all these emerging enablers, notable barriers remain for the widespread adoption of these opportunities within the U.S. nuclear fleet, including the following:

- Assurance of the markets for alternate products needed to support decision-making for large capital modification investments.
- Electric utility mindset and business history centered solely on producing electricity.
- Design change complexity and regulatory uncertainty associated with plant modifications needed to support alternate product streams.

The DOE Light Water Reactor Sustainability (LWRS) Flexible Plant Operations and Generation (FPOG) Pathway is developing options to help U.S. nuclear power plants (NPP) in all these areas to enable NPPs designed for steady baseload operation to integrate with intermittent wind and solar capacity to assure reliable clean energy for the nation. Current and near-term laboratory research is focusing on the technical, regulatory, safety, demonstration, and economic elements in support of improving nuclear plant flexibility through hybrid production of electricity and other non-electric products such as hydrogen and energy arbitrage.

One key research area required to validate the feasibility of hydrogen production at NPPs is related to how supporting design changes would conform to the regulatory requirements of the U.S. Nuclear Regulatory Commission (NRC). Design changes are routinely performed at operating U.S. nuclear power reactors using two parts of Title 10 of the Code of Federal Regulations (CFR). If a design change can be shown to present little change in safety from the current licensing basis, the licensee can implement the change without prior NRC approval under 10 CFR, Part 50.59. When a proposed change to the facility is determined not to be within the limits specified in 10 CFR 50.59, the licensee must gain NRC approval before implementation using the license amendment process described in 10 CFR 50.90.

The Hydrogen Regulatory Research Review Group (H3RG) was formed to generically research the magnitude of high temperature hydrogen electrolyzer technology addition at NPP's that could potentially be accepted under a 10 CFR 50.59 evaluation. Knowledge of this threshold would help individual licensees to understand the significant contributors to that evaluation and help them focus their design evaluations on the most important risk contributors. A design that can be approved under a 10 CFR 50.59 evaluation is desired to minimize unnecessary licensee engagement in the NRC's license amendment request (LAR) process.

Several plant-specific licensed design elements were identified early in the process (Section 2.2.2 Sub-Committee Specific Considerations) where additional site-specific licensing-related evaluations are expected to be required. Thus, the generic guidance described by this research is targeted at reducing complex regulatory approvals under the LAR process that might otherwise be required. In support of this

objective, the H3RG includes a broad collaboration with primary participants from DOE-supported national laboratory research leads, contracted architect engineering (AE) participants, as well as nuclear plant operators, and expertise from licensing and design disciplines.

All research and development (R&D) described by revisions to this report is based on the underlying premise of three key research support elements as shown in Figure 1.

Figure 1. R&D approach to nuclear integrated high temperature electrolysis regulatory approval

Over the report revisions [1], these R&D components have continued to mature across progressive plant integration designs, regulatory evaluation development, and rigorous plant risk analyses.

- Revision 0 (April 2022):
  - Documented early technical and regulatory research findings associated with the pairing of an assumed 1200 MW_e generic pressurized water reactor (GPWR) design with an integrated 100 MW_{nom} nuclear electric/steam-powered hydrogen electrolysis design.
  - Evaluated high-temperature steam electrolysis (HTE) as the paired hydrogen-generating technology design case based on use of both clean electric and clean steam to achieve higher efficiencies than electric-only low-temperature electrolysis (LTE).
  - Provided the first simplified 100 MW_e and 25 MW_{th} electric supply/steam extraction conceptual design basis for a nuclear integrated hydrogen plant by HTE and evaluated the plant operational responses under various HTE facility operating modes.
  - Performed a simplified 6-question 10 CFR 50.59 evaluation.
  - Evaluated preliminary risk bases in support of the integrated nuclear/hydrogen design.
  - Demonstrated early supporting design and regulatory approval path elements.
- Revision 1 (Nov 2022):
  - Further evaluated and explained subsequent regulatory research findings with specific emphasis on the likely degree to which the simplified 10 CFR 50.59 process could be used.
  - Regulatory R&D-based conclusions were provided in the form of a generic 10 CFR 50.59 evaluation addressing all eight questions for a 100 MW_e and 25 MW_{th} electric supply/steam extraction conceptual design.
  - Identified site-specific licensing requirement areas that will likely require evaluation outside the 10 CFR 50.59 process.
- Revision 2 (Aug 2023):
  - Documents conceptual design aspects supporting the integration of an expanded nuclear electric/steam-powered hydrogen electrolysis designs between 100 MW_{nom} to 500 $MW_{nom.}$
  - Describes the successful 10 CFR 50.59 evaluation basis for a 500 MW_{nom} design case [9] and provides caveats for plant specific licensing considerations that may need to be addressed outside the 10 CFR 50.59 process.
  - Incorporated hazard analyses, sensitivity studies, and other analysis refinements since the last revision to this report that support the placement of (100, 500, and 1000 MW_{nom}) High Temperature Electrolysis Hydrogen Production Facilities (HTEF) co-located with a NPP

("Expansion of Hazards Analysis and Probabilistic Risk Assessments of a Light-Water Reactor Coupled with Electrolysis Hydrogen Production Plants" [14]). This also included refinements to the calculated safe distance placement of the HTEF and other analyses that inform the overall probabilistic risk assessment (PRA) of the NPP. Both the deterministic and probabilistic results help support the licensing case for the proposed changes to the NPP and safe siting distance of the HTEF. Specific results from that analysis work [14] contributing to Revision 2 to this report include:

- More detailed specification of the hydrogen electrolysis facility design, footprint, and safe distance layout of 100, 500, and 1000 MW_{nom} design cases.
- HyRAM++ detonation analysis conclusions from rigorous plant siting and failure modes and effects analysis input to confirm hydrogen plant standoff requirements.
- Inclusion of specific details for the range of heat extraction system design analysis elements used in the PRA model with results supporting minimal increases to core damage frequency and large early release frequency per Regulatory Guide 1.174.
- Risk consideration beyond the quantified risk assessment for the NPP, including qualitative hazards assessment for the community and potential effects on utility business risk.

These enhanced realism and associated risk analysis refinements further reduced the calculated required stand-off distance from a previous value of 500 meters to as close as 200 meters from the nuclear power plant's transmission towers which have been identified as the most critical susceptible structure, system, or components [12].

It is noted that although Reference [14] also simplistically provided hazard evaluation for a 1000 MW_{nom} case, it is not yet supported by pre-conceptual design development and 10 CFR 50.59 evaluation rigor comparable to that completed for the 100–500 MW_{nom} cases and as such is provided for information only where referenced in this revision.

Revision 2 also provides a description of the major supporting research deliverables as a guide for industry licensees and architect engineers that may be developing plant-specific design and regulatory approaches.

It should be noted that this report (and its previous revisions) does not specify the degree to which licensees will be able to use a site-specific 10 CFR 50.59 evaluation to demonstrate conformance with their specific NPP's design and current licensing basis. Such plant-specific evaluations are beyond the current scope of this report.

The design, risk assessment, and 10 CFR 50.59 evaluation R&D research results described in this revision further contribute to the body of technical and regulatory knowledge intended to support the success of streamlined licensee evaluation of design and regulatory aspects associated with the cogeneration of nuclear integrated hydrogen by HTE.

Subsequent revision of this report is planned to document future laboratory research support of DOE co-funded Hydrogen HUBb projects involving nuclear integrated hydrogen by HTE - which had not sufficiently progressed to include in Revision 2 as of the writing of this report.

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CONTENTS

FIGURES

TABLES

ACRONYMS

AE	architectural engineer
B&W	Babcock & Wilcox
BIL	Bipartisan Infrastructure Act
BWR	boiling water reactor
CAFTA	Computer Aided Fault Tree Analysis
CDF	core damage frequency
CFR	Code of Federal Regulations
CLB	current licensing basis
DBA	design basis accident
DCS	digital control system
DOE	Department of Energy
DOT	Department of Transportation
FEED	front-end engineering and design
FMEA	failure modes and effects analysis
FPOG	flexible plant operations and generation
FSAR	final safety analysis report
GPWR	generic pressurized water reactor
H3RG	Hydrogen Regulatory Research Review Group
HELB	High Energy Line Break
HES	heat extraction system
HFT	heat transfer fluids
HFTO	Hydrogen and Fuel Cell Technologies Office
HP	high pressure
HTE	high temperature electrolysis
HTEF	high-temperature electrolysis facility (hydrogen island)
HTSE	high-temperature steam electrolysis
HyRAM+	Hydrogen Plus Other Alternative Fuels Risk Assessment Models
I&C	instrumentation and control
IAEA	International Atomic Energy Agency
IES	Integrated Energy Systems
iFOA	Industrial Funding Opportunity Announcement
IIJA	Infrastructure Investment and Jobs Act
INL	Idaho National Laboratory

IP	Indian Point Energy Center
IRA	Inflation Reduction Act
kW	kilowatt
LAR	License Amendment Request
LBE	licensing basis events
LERF	large early release frequency
LOCA	loss-of-coolant accident
LOOP	loss-of-offsite-power
LOOP-SW	switchyard-based LOOP
LP	low pressure
LTE	low-temperature electrolysis
LWR	light water reactor
LWRS	Light Water Reactor Sustainability
MCR	main plant control room
MSIV	main steam isolation valves
MSLB	main steam line break
MSR	moisture separator reheater
MW_e	megawatt electrical rating (electrical power)
MW_{nom}	megawatt (nominal hydrogen plant electrical powering requirement)
MW_{th}	megawatt thermal rating (thermal power)
NERC	North American Electric Reliability Corporation
NPP	nuclear power plant
NRC	Nuclear Regulatory Commission
P&ID	piping and instrumentation diagram
PA	protected area
PHMSA	Pipeline and Hazardous Materials Safety Administration
PRA	probabilistic risk assessment
PWR	pressurized water reactor
R&D	research and development
RIPE	Risk Informed Process for Evaluation
S&L	Sargent & Lundy
SME	subject matter experts
SNL	Sandia National Laboratories
SSC	system structure or component
TNT	trinitrotoluene

TPD	thermal power delivery
TS	technical specifications
UFSAR	updated final safety analysis report
VTO	Vehicles Technologies Office

REPORT ON THE CREATION AND PROGRESS OF THE HYDROGEN REGULATORY RESEARCH REVIEW GROUP

1. INTRODUCTION

1.1 Background: Why Nuclear Generated Hydrogen

The emerging gap between the growth of non-dispatchable renewable energy generation and lagging clean energy storage contributes to the expansion of excess clean generation on the grid. The overlapping impact of the dominant clean generating sources (intermittent renewables and baseload nuclear power) exacerbates this challenge during parts of the daily supply-and-demand cycles.

A contributing factor is that both intermittent renewables and baseload nuclear have inherent flexibility constraints in their operational models. Nuclear power has significant near-term potential to change its long-standing operational model by shifting generation output away from electrical generation when there is not additional grid demand for clean energy. During these times, nuclear could flexibly produce real-time usable or storable clean energy to decarbonize functions across the power, industrial, and transportation sectors. Specifically, hydrogen by electrolysis has the potential to favorably influence these sectors as a storage medium and energy carrier for excess intermittent carbon-free generation.

In recent years, the development of water-splitting electrolysis systems has dramatically accelerated as the interest has increased in clean hydrogen production and global decarbonization of transportation, industry, and other sectors. Electrolyzed hydrogen produced by renewables and low-temperature electrolysis (LTE) is already emerging as a near-term clean stored-energy carrier. This clean storage capability will be an important complement to limited renewable electricity storage via Lithium-Ion batteries and other emerging storage technologies. HTE systems can achieve higher overall system efficiencies compared to LTE. Nuclear generators are unique in their capability to deliver both clean electrical and heat energy output—the two components needed to produce hydrogen by clean, high-efficiency HTE, shown in Figure 2.

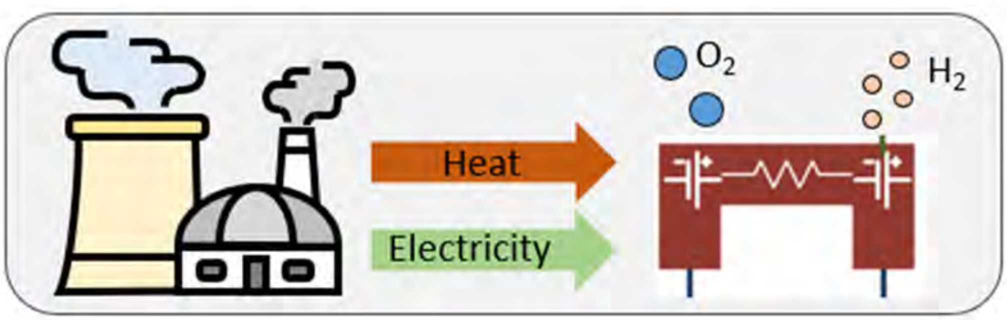

Figure 2. Nuclear provides heat and electricity for high-temperature electrolysis.

DOE support under the LWRS FPOG Pathway at INL is accelerating key technology development in this area. The current LWRS R&D focus regarding implementation of integrated hydrogen generation at nuclear facilities is addressed through exploring practical pre-conceptual designs, pilot hydrogen projects, and development of licensing success paths consistent with the United States Nuclear Regulatory Commission (NRC) requirements. This licensing R&D review element continues to be developed by nuclear industry design and regulatory experts under the H3RG.

2. LABORATORY AND INDUSTRY REGULATORY COLLABORATION

2.1 Design and Regulatory Evaluation Approach

2.1.1 Demonstration Design

Revisions 0 and 1 to this report provided the pre-conceptual design elements necessary to support the linkage of a medium-scale 100 MW_{nom} High-Temperature Electrolysis Facility (HTEF) to an assumed 1200 MW_e generic pressurized water reactor (GPWR) nuclear power plant (NPP) design. Revision 2 includes recently issued design, regulatory and cost basis information for further scaling up this evaluated hydrogen plant size to 500 MW_{nom} [9].

This collective research work through Revision 2 of this report provides technical and licensing bases for a scaled-up DOE collaboration pilot project on nuclear integrated HTE beyond current industry hydrogen demonstration projects underway at several U.S. nuclear utilities. In comparison to these small-scale kW-level pilots, nuclear HTE encompassing analyzed MW_e-level research cases as shown in Table 1 is approximately an order of magnitude larger. This range of designs based on both NPP electrical and heat energy introduces more complex plant equipment integration, operational interaction, and regulatory considerations for evaluation. The 100–500 MW_{nom} scale pilot project evaluation addressed in [1 and 9] provided conceptual thermohydraulic, electrical, and controls integration design basis analyses as summarized in Table 1. From a hazard analysis standpoint, Reference [14] also simplistically evaluated a 1000 MW_{nom} case, however as it has yet to be supported with the same design and 10 CFR 50.59 evaluation rigor as provided for the 100–500 MW_{nom} cases it is noted here for information only.

Table 1. Summary HTEF design parameters by plant size [9].

Parameter	Unit	HTEF Size	
		100 MW_{nom}	500 MW_{nom}
Hydrogen Production Capacity[a]	U.S. tons/day	55–58	275–290
H_2 Plant Electric Load MW_e	MW_e	105	500
H_2 Plant Auxiliary Loads + Margin	MW_e	22	50
Power Factor	—	0.92	0.92
Total Electrical Power Requirements	MVA	140	600
H_2 Plant Thermal Load	MW_{th}	20	100
Plant Thermal Losses + Margin	MW_{th}	5	5
Total Thermal Power Requirements	MW_{th}	25	105
Steam Input Temperature[b]	°F	>300 (333)	>300 (333)
Steam Input Pressure[b]	psig	>50 (59.3)	>50 (59.3)
Separation Distance from PWR	m	250 & 500 m	250 & 500 m

a Conservative production capacities are shown based on 2022 values. Technology improvements over the next few years are expected to improve the yields of these plants to approximately 60 and 300 U.S. tons/day, respectively.

b Parameters in parenthesis indicate actual design values.

Figure 3 and Figure 4 provide graphic representations of the planned progression of nuclear-integrated HTE demonstration projects from small- to large scale.

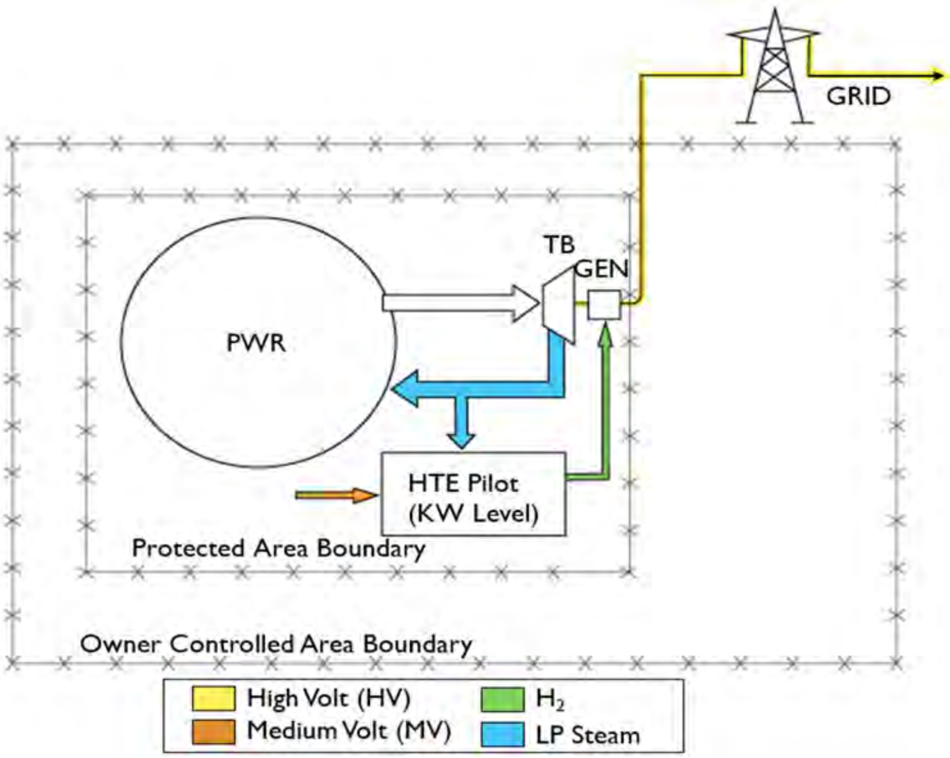

Figure 3. Early 2020s – Small kW-scale HTE demonstrations.

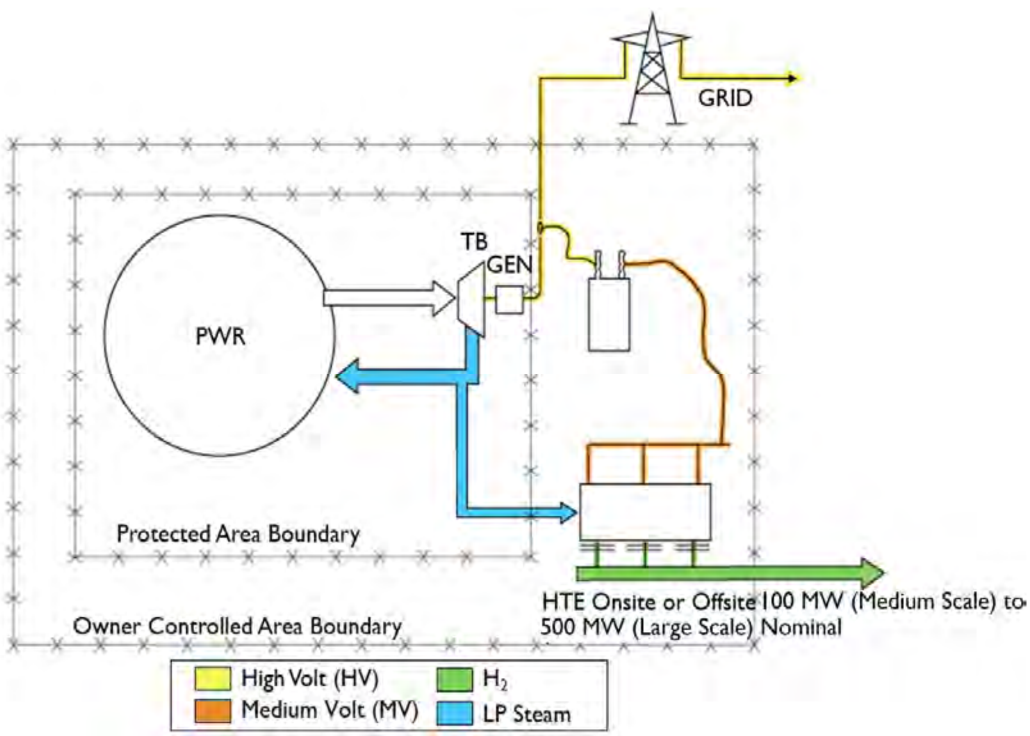

Figure 4. Late 2020s – Medium-to-large MW-scale HTE demonstrations.

2.1.2 HTEF Linkage and Safety Analysis

All light water reactors already have onsite small hydrogen storage facilities to support plant processes including main generator cooling. Because nuclear integrated HTE introduces additional hydrogen gas volumes associated with production, distribution, compression, and storage of hydrogen near the NPP, a collection of hazard analyses that support the placement of a HTEF co-located with a NPP was recently issued ("Hazards and Probabilistic Risk Assessments of a Light Water Reactor Coupled with a High-Temperature Electrolysis Hydrogen Production Plant" [14]). This report further informed approaches originally determined under "Probabilistic Risk Assessment of a Light Water Reactor Coupled with a High-Temperature Electrolysis Hydrogen Production Plant." [3] to determine calculated safe distance placement of the HTEF and provide additional analyses input to the probabilistic risk assessment (PRA) of the NPP.

A fundamental premise of the body of work circumscribed by the H3RG, and laboratory supported design, modeling, testing, and hazard analyses is that both deterministic and probabilistic research insights are needed to support the licensing case for proposed coupling of a NPP with HTE and the related siting (standoff) distance between the two. Notably, enhanced realism and associated risk analysis refinements included in [14] support an allowable stand-off distance reduction from the previously analyzed 500 meters [3&8] to as little as 200 meters from the nuclear power plant's transmission towers which have been identified as the most critical susceptible structure, system, or component [12]. Specific supporting research elements included:

- More detailed specification of the hydrogen electrolysis facility design, footprint, and safe distance layout of the 100–500 MW_{nom} design cases
- HyRAM++ detonation analysis conclusions from rigorous plant siting and failure modes and effects analysis input to confirm hydrogen plant standoff requirements
- Inclusion of specific details for the range of heat extraction system designs analysis elements used in the PRA model with results supporting minimal increases to core damage frequency and large early release frequency per Regulatory Guide (RG) 1.174
- Risk consideration beyond the quantified risk assessment for the NPP, including qualitative hazards assessment for the community and potential effects on utility business risk.

Another aspect of the evaluation was to present an initial estimate of the impact on NPP risk attributed to the coupling of nominal HTEF designs (100, 500, and 1000 MW_{nom}) for consideration in the NPP PRA. Specific assessment areas thus included design basis initiating event frequencies, core damage frequency (CDF) and large early release frequency (LERF). In addition to an assumed loss of offsite power (LOOP) caused by damage to the switchyard, the failure scenarios evaluated impact on CDF through damage to site safety-related structures, systems, and components (SSCs). The base assumption of the PRA models was that the HTEF was located 1 km from critical NPP components. This was based on sensitivity studies conducted to determine the minimum safe distance. Revision 1 to [3] concluded that:

- NPP modifications include equipment that must be evaluated risk in the PRA related to:
  - Extraction of thermal energy from the secondary system steam downstream of the high-pressure turbine to a heat extraction system (HES)
  - The upstream NPP effects of heat energy demand across the HES reboiler isolation boundary supplying vaporized water to the HTEF electrolyzers.
- Two metrics were evaluated for effects on initiating event frequencies, CDF and LERF [3]. The largest increase in initiating event frequency resulting from the addition of the HES occurred for a main steam line break (5.6%) which is considered minimal in plant modification licensing decisions [3]. In addition, two PRA metrics were evaluated for overall increase in CDF and LERF.

- [3] for NRC Reg Guide 1.174 support and results were in the most acceptable ranges.
- All HES additions for the range of HTEFs considered caused a maximum increase of 1.3% in LOOP-SW initiating event frequencies, provided that proper shielding is in place for the 500 MW_{nom} and 1000 MW_{nom} combined hydrogen production headers leaving the HTEFs. The leak masses assumed in the PRA were very conservative (a 100% pipe rupture). The analysis concluded that if more realistic leak assumptions are applied in the future, the larger HTEFs could be located in closer proximity to the NPP without engineered barriers.
    - The report specifically concluded that an HTEF could safely be located at distances approaching.2 km from these components [1&2] for the for the 100 MW_{nom} HTEF without engineered barriers in place. Both the 500 MW_{nom} and 1000 MW_{nom} HTEFs could also potentially use this same.2 km standoff if their combined final stage hydrogen production headers are buried or employ other engineered barriers. If the combined final stage header is not protected, their standoff distances would increase but are expected to remain within 1 km. This report and the supporting generic 10 CFR 50.59 evaluation assumes the HTEF is located at a standoff distance of 0.5 km. Actual safe distance could be less or more as described above depending on results of future site-specific analysis expected to be performed by plants considering adding an HTEF. Site-specific analysis was beyond the scope of this generic evaluation. However, the typical guidance methodology referenced herein could be employed for appropriate siting of an HTEF in proximity to an existing NPP.

A new PRA has been completed [14] and will be included as a successor to the laboratory report [3]. The new bounding hydrogen detonation-caused initiating event was a LOOP. The supporting PRA analysis also includes significant enhancements in realism over the previous research. For example, even though sequential cascading HTEF module failures were assumed the conservatively assumed bounding accident was from full leakage of hydrogen from one of the electrolysis modules due to each producing the same repetitive detonation effect. Therefore, the size of the facility did not materially affect the overpressure potential, only the frequency with which the event occurred. The hazard analyses, sensitivity studies, and other analysis refinements incorporated since the last revision to this report include:

- Details for the range of heat extraction system designs analyzed in the PRA model [14] which yielded results showing minimal increase in CDF and LERF per RG 1.174 [13].
- More detailed specification of the HTEF design, footprint, and safe distance standoff for 100, 500, and 1000 MW_{nom} design cases.
- A specific licensing support initiative directed by the H3RG which demonstrated an analytical tool to quantify the enveloping effects of hydrogen release and detonation from realistic failure modes and effects involving HTE plants. The intent of this new R&D element was to support NPPs that were not previously licensed for nearby explosions and thus may need to provide plant-specific evaluation of hydrogen detonation impacts and site-specific standoffs to preclude the possibility for an accident of a different type than previously evaluated in the Final Safety Analysis Report (UFSAR).

In support of this initiative, Hydrogen Plus Other Alternative Fuels Risk Assessment Models (HyRAM+) Bauwens-Durofeev (Bauwens) hydrogen jet leak detonation overpressure analyses were developed for the 100, 500, and 1000 MW_{nom} HTEF designs. The capabilities pedigree of the HyRAM+ open-source tool are described as follows in [10]:

> *"A software toolkit that integrates publicly available data and models relevant to assessing the safety in the use, delivery, and storage infrastructure of hydrogen and other alternative fuels (i.e., methane and propane). The HyRAM+ risk assessment calculations incorporate probabilities of equipment failures for different components for both compressed gaseous and liquefied fuels, and probabilistic models for the effect of heat flux and overpressure on people.*

HyRAM+ also incorporates experimentally-validated models of various aspects of release behavior and flame physics. The HyRAM++ toolkit can be used to support multiple types of analysis, including code and standards development, safety basis development, facility safety planning, and stakeholder engagement.

HyRAM++ was supported by the U.S. Department of Energy (DOE) Office of Energy Efficiency (EERE) Hydrogen and Fuel Cell Technologies Office (HFTO), the DOE EERE Vehicles Technologies Office (VTO), and the U.S. Department of Transportation (DOT) Pipeline and Hazardous Materials Safety Administration (PHMSA)."

- A detonation analysis approach which included rigorous HTEF plant layout dimensioning and Failure Modes and Effects Analysis (FMEA) input to confirm hydrogen plant stand-off requirements [7].
- Specific hydrogen detonation fragility evaluation of SSCs that are important to plant safety including these:
  - All Category I Structures
  - Storage Tanks (CST, RWST, etc.)
  - Circulating Water/Service Water Pump Area
  - Standby Auxiliary Transformer
  - Switchyard
  - General Transmission Tower.
- Risk considerations beyond the quantified risk assessment for the nuclear power plant including qualitative hazards assessment for the local environs of the model plant and potential effects on utility business risk.

In summary, the HTEF designs modeled for the most recent PRA revision [14] has replaced excessively conservative assumptions from the prior Revisions 0 and 1 with a PRA [3] in this revision. These results further support the feasibility of reducing the calculated stand-off distance for the high-pressure compression stage components of the assumed 100, 500, and 1000 MW_{nom} HTEFs from a previous value of 500 meters to roughly 200 meters from the nuclear power plant's transmission towers (the most susceptible SSC) [12]. Figure 5 and Figure 6 provide visual representation of HTEF layout and plant stand-off concept used in [3 and 9].

Figure 7 shows representative individual low pressure HTE module under test at INL. An array of these modules and the attendant system interface equipment (e.g. HES, electrical power transformers and distribution, etc.) collectively makes up the HTEF. Additionally, these results confirm the intent of 10 CFR 50.59, Question 1 - that the HTEF does not result in more than a minimal increase in initiating event frequencies for all design basis accidents, with none exceeding 6% for any of the cases proposed. Also see [9] for 500 MW nom 50.59 generic evaluation details. Further, the PRA [14] results for CDF and LERF are consistent with RG 1.174 [13].

These PRA analysis results (and supporting assumptions) are discussed in greater detail in the recently issued PRA [14].] The results include details for nuclear plant SSCs based on fragility analysis results recently issued by Sandia National Laboratory [3] which summarize the results of hydrogen detonation analyses derived by detailed failure modes and effects analysis [6 and 7] with methodology as described in [11].

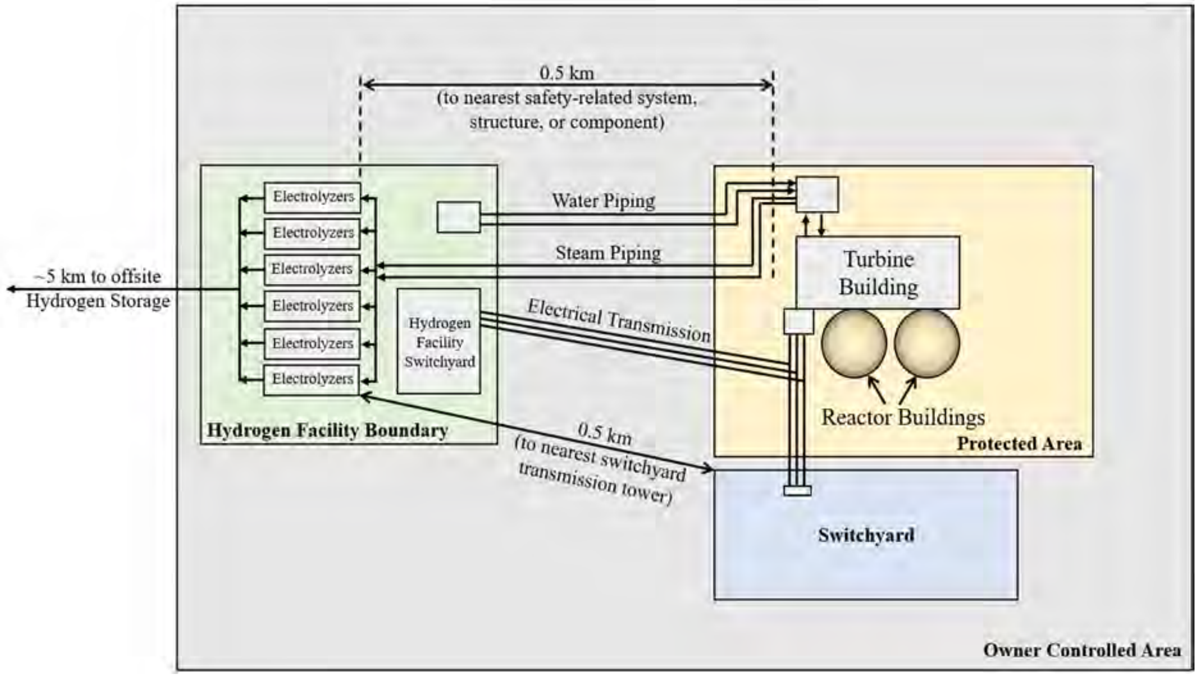

Figure 5. HTEF layout and plant standoff concept.

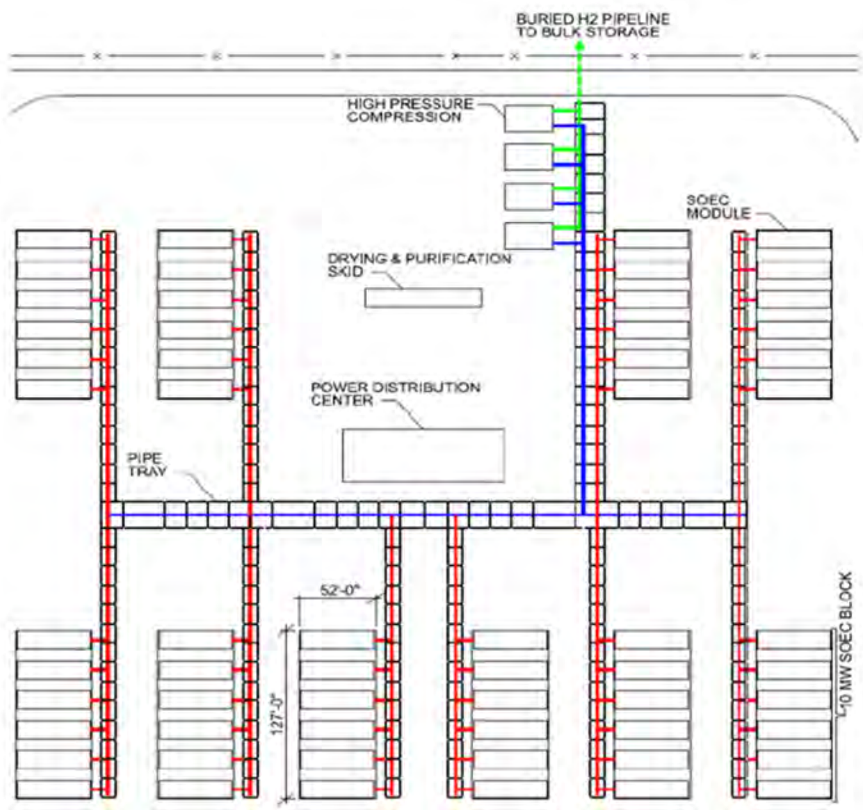

Figure 6. 100 MW Hydrogen Facility layout.

Figure 7. Low pressure 100 kW HTE subcomponent module installed at INL.

2.1.3 Regulatory Considerations

Recent Progress and Actions

The conceptual 100 MW_{nom} to 500 MW_{nom} design package issued under [9] further advances the path to HTE at scale. This includes the regulatory approval path supported by related emerging reports referenced in this revision [3, 6, 7, and 14] which serve as resources for licensees to evaluate plant-specific design and regulatory approval strategies. Next-step plant-specific licensee evaluation of MW-level conceptual design and HTEF integration aspects will be critical in demonstrating the extent to which the 10 CFR 50.59 evaluation process can be employed or whether submittal for NRC approval under a license amendment process may be needed. Laboratory and H3RG support for licensee plant-specific regulatory approvals and supporting interactions with NRC will thus remain important emerging research support areas as described in Sections 2.4.1 and 2.4.2.

Background

As described in Revision 0 to this report, use of the 10 CFR 50.59 evaluation process is allowed only after determining that a change to the plant's Technical Specifications (TS) is not necessary and then a proposed change to the facility can be implemented without prior NRC approval if it satisfies the following criteria:

1. Does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report (as updated)
2. Does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the final safety analysis report (as updated)
3. Does not result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report (as updated)

4. Does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the final safety analysis report (as updated)
5. Does not create a possibility for an accident of a different type than any previously evaluated in the final safety analysis report (as updated)
6. Does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the final safety analysis report (as updated)
7. Does not result in a design basis limit for a fission product barrier as described in the UFSAR (as updated) being exceeded or altered
8. Does not result in a departure from a method of evaluation described in the UFSAR (as updated) used in establishing the design bases or in the safety analyses.

If any of the above criteria are not met, then the 10 CFR 50.59 process cannot be used to implement the modification, and a license amendment request (LAR) must be submitted to the NRC for approval prior to implementation. The licensee is required periodically to submit to the NRC a list of all 10 CFR 50.59 evaluations that have been completed. This gives the NRC a *post facto* opportunity to review plant modifications justified under 10 CFR 50.59.

The intent of collective regulatory research elements being developed by laboratory R&D, contracted architectural engineering (AE), and nuclear industry regulatory experts is to explore utility adoption strategies for HTE designs within the current licensing basis where feasible.

With that intent in mind, the H3RG was established based on input from INL-led FPOG stakeholder meetings under several industry subcommittee research areas with leads selected from industry expert design, operational, and regulatory areas. The regulatory input path to the current generic 10 CFR 50.59 products is informed by the collaborative efforts of these expert participants in the form of industry advisory subcommittees and primary/secondary interfacing organizations as visually represented in Figure 8. Targeted use of expert review resources allowed for early identification of discrete subcommittee research areas that have informed design and regulatory R&D efforts.

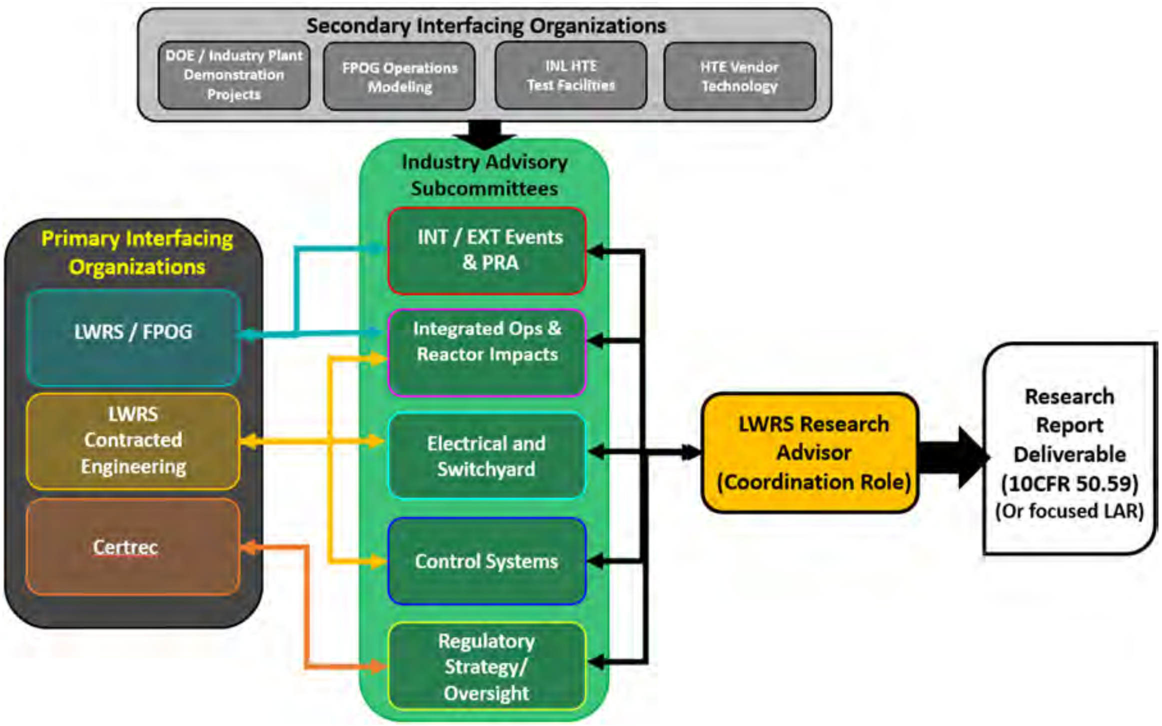

Figure 8. H3RG organizational and subcommittee structure.

During the H3RG formation stage, an experienced INL Program adviser was contracted from the nuclear industry. This brought in industry perspectives on large project design and implementation as well as experience in obtaining regulatory approval for complex design changes via 10 CFR 50.59 site evaluations and formal NRC LARs. This led to the establishment of foundational design and licensing considerations for a comprehensive regulatory evaluation approach including:

- Steam line connections and mass steam flow for operational and faulted conditions
- Consideration of steam leak assumptions on existing plant analyses
- Secondary plant dynamics and operator control issues
- Analog and digital control schemes and limits of manual control, including human system dynamics
- Operational considerations related to thermal energy extraction, including any effects on the reactor core
- Dispatch limitations and transitions between electrical and hydrogen production
- Electrical system design interactions and power off-take dynamics
- Hydrogen equipment physical plant standoff requirements and onsite storage limits based on detonation analysis design requirements
- Plant PRA considerations – Increases in frequencies of design basis event initiating events, CDF, and LERF
- Licensing and design basis events compatibility.

The H3RG also established the working assumption that the HTE design would be under site operational control to ensure that system operation would have no unintended impacts on nuclear reactor power not directly under operator control. It was also recognized that this critical station operational/reactivity control philosophy must be adhered to independent of whether the HTEF end-use hydrogen generation equipment was established under a utility or vendor operated business model. As of the issuance of Revision 2 to this report, no specific nuclear HTE project has been announced that defines intended operational authority boundaries between a NPP and an HTEF. The operational authority issue will need to be evaluated as part of future nuclear integrated hydrogen projects under 10 CFR 50.59 evaluation to assure that no uncontrolled reactivity impacts are introduced through plant/vendor operational control issues. As a point of reference, the currently envisioned 100 MW_{nom} to 500 MW_{nom} designs effectively limit the NPP control room operational interface with extraction feed to the NPP/HTEF reboiler interface to that of a “steam supply on" or “steam supply off" function [9].

Another premise for the H3RG was that a design change to implement nuclear integrated hydrogen by HTE must be screened for effects on the existing facility and procedures as described in the UFSAR, as well as the licensing bases, and that a formal 10 CFR 50.59 evaluation will be required in all MW-level design cases. To inform the use of the 10 CFR 50.59 process for an HTE evaluation, the project relied on expert review for:

- Comparative reviews of industry examples where changes to the facility were appropriately evaluated under 10 CFR 50.59 – especially for first-of-a-kind changes and deliberate operating premise shifts.
- Detailed reviews of past NRC feedback on use of 10 CFR 50.59 and lessons learned on the limits of use of the 10 CFR 50.59 process for approving changes to the facility.
- Review of ongoing industry 10 CFR 50.59 evaluations issued in support of LTE modifications or small-scale (kW level) HTE demonstrations.
- Consideration of historical regulatory challenges related to combustible gas concerns at nuclear facilities.

During an H3RG meeting held on 8-17-23, feedback was received from NRC regarding the importance of rigorously considering Indian Point Energy Center (IP) operating experience regarding licensee/regulatory evaluation and approval interactions for the acceptance of a natural gas line traversing the Owner Controlled Area of the NPP. The discussion reinforced the need for the H3RG to further evaluate and where applicable, adopt lessons learned in the areas of analytical rigor, plant equipment effects, and public perception. Background material on this operation experience [18] was previously considered by the H3RG in an earlier research phase, but additional follow-on review of these lessons learned will be considered with the intent of being able to discuss a comparative review of the Indian Point regulatory approval and laboratory research approaches and deliverables now reaching intended maturity. This is captured for further action under Section 2.4.2.

The multifaceted laboratory, AE, and industry expert collaboration framework described above informed the industry-first MW-level conceptual nuclear integrated HTE design under Reference [1]. Laboratory-contracted AE, Sargent & Lundy (S&L), has brought extensive expertise in complex design and regulatory evaluation which significantly contributed to the collaboration. Westinghouse also made significant contributions to the instrument and control design aspects necessary to couple an NPP to an HTEF without reactivity feedback to the NPP [1and 9]. CERTREC Corporation was contracted to manage the organization and coordination of the H3RG industry group based on its extensive nuclear experience with facilitation of complex regulatory interactions with the NRC.

Revisions 0 and 1 to this report addressed “site-specific” regulatory considerations that represent typical U.S. nuclear plant licensing challenges and noted that site-specific considerations identified during H3RG reviews would not be evaluated in detail as part of the scope of these generic 10 CFR 50.59 evaluation deliverables. This was determined to be an issue of practicality based on the wide range of plant-specific licensing basis nuances.

As H3RG review progressed during the design review phase, it was recognized that unique site circumstances and variety of licensing basis elements across the U.S. nuclear fleet made it impractical to develop an enveloping set of licensing requirements to include in a ready-for-use draft 10 CFR 50.59 product. It was thus determined that typical site-specific features would be identified and captured for future use by nuclear utility demonstration or full-scale implementation designers and regulatory staff. Those site-specific areas will help build FEED study considerations to characterize the scope of site-specific licensing requirements to be addressed. Site-specific considerations identified during the review are documented in Section 2.2 and Appendix C and the CERTREC web-based work platform used by the H3RG. Appendix C makes appropriate ties to individual subcommittee considerations that may be of value to future FEED study developers. These include many, but not all site-specific review areas that must be considered.

Although the intent of ongoing regulatory research is to provide a utility basis for integration of HTE within the bounds of 10 CFR 50.59, other regulatory approval paths are possible. These could fall within two main areas:

1. Large-Scale Integration Complexities

   Although the early stages of the R&D evaluation of the proposed plant modifications for conceptual generic design of the 100 MW_{nom} HTEF did not identify conflicting regulatory areas that would preclude the use of the 10 CFR 50.59 process, concerns were raised that the planned larger scale 500MW_{nom} design might introduce limitations on the use of the 10 CFR 50.59 evaluation process. This was not found with the exception of plants without original licenses that evaluated the effects of explosion risks in proximity to the NPP [9]. This potential original licensing limitation likely applies regardless of sizing across medium-scale nuclear integrated HTEF designs.

2. Plant-Specific License Requirements

As previously discussed, the H3RG has identified that even when new HTE design elements align with use of the 10 CFR 50.59 process, other unrelated plant-specific license requirements may lead to additional required approvals from the NRC. The current design and regulatory evaluation work in support of the medium-scale 100 MW_{nom} to 500 MW_{nom} demonstration design cannot address the full spectrum of plant-specific licensing aspects. It is expected that such considerations will require NRC approval that will naturally emerge as utility FEED studies progress. These are expected to be addressed based on HTE projects that are selected:

- Through DOE-sponsored activities like nuclear hydrogen hubs
- As part of Industrial Funding Opportunity Announcement (iFOA) awards
- During utility design change development outside those award processes.

Section 2.4 addresses furthering ongoing NRC and industry discussions [15 & 16] around consideration for the use of licensee risk-informed evaluations to seek streamlined approval of certain low risk nuclear integrated HTE integration design issues including specific changes for which probabilistic analyses demonstrate that the changes are of very low safety significance [17].

The next phase of industry and laboratory supported engineering, licensing, and economic evaluations could involve participation in detailed FEED studies to identify such plant-specific considerations. This would support moving forward on nuclear integrated HTE adoption while minimizing potentially repetitive utility-required design and regulatory support products. This could be particularly useful for evaluating commonalities that may not meet the acceptance criteria for a 10 CFR 50.59 evaluation.

Even in HTE situations where specific NRC approval may be required, future laboratory research deliverables could provide a basis for meeting the intent of the regulatory requirements. One example of this would be the use of existing laboratory hydrogen detonation analysis tools for sites that cannot meet the hydrogen island siting assumptions outside of the Protected Area (PA) and/or that have a PA boundary located closer to safety-related SSCs than supported by RG 1.91 [4] stand-off requirements.

2.2 Summary Regulatory Findings and Considerations

2.2.1 Generic 10 CFR 50.59 Summary

The first phase of the 10 CFR 50.59 evaluation draft (Reference [1]) completed by S&L provided a preliminarily assessment of six of the 10 CFR 50.59 criteria. This comprised a pre-screening review to evaluate the regulatory feasibility of the proposed coupling of a NPP to an HTEF as listed below:

1. Frequency of occurrence of an accident previously evaluated in the UFSAR
2. Likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR
3. Radiological consequences of an accident previously evaluated in the UFSAR
4. Possibility for an accident of a different type than any previously evaluated in the UFSAR
5. Possibility for a malfunction with a different result than any previously evaluated in the UFSAR
6. Departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

The preliminary assessment of the six critical questions was reviewed and discussed with the H3RG. This suggested that utilities could be successful using the 50.59 evaluation process. Utilizing the H3RG feedback, S&L performed a detailed review of all eight of the 10 CFR 50.59 questions and completed the generic 50.59 evaluation (documented in Revision 1 of this report). This generic 10 CFR 50.59 (c)(2) evaluation was subsequently re-evaluated against the current demonstration design for the acceptability of implementing up to a 500 MW_{nom} modification as a proposed change to an assumed 1200 MWe GPWR [9]. This evaluation was introduced in Revision 2 of this report and represents the most recent generic evaluation guide for use by industry stakeholders considering the proposed coupling of a NPP to a hydrogen production facility of any size up to 500 MW_{nom}.

Plant modifications for integrating the HTEF into the nuclear power plant have the potential to increase the frequency of occurrence of accidents previously evaluated in the UFSAR. The accidents that are of potential interest are:

- Excessive Increase in Secondary Steam Flow
- Loss of External Electrical Load
- Loss of Offsite Power
- High-Energy Line Break
- Flooding
- Turbine Trip

The potential impacts to the accidents previously evaluated are discussed below.

Excessive Increase in Secondary Steam Flow

The diversion of steam from the cold reheat piping to the new reboiler introduces the potential for a transient condition in the steam systems. This has the potential to affect the frequency of occurrence of an Excessive Increase in Secondary Steam Flow.

An excessive increase in secondary system steam flow (or excessive steam flow event) is defined as a rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand.

To accommodate routine operations at the nuclear plant, the design basis for normal operation of the plant control systems includes 10% step changes and 5% per minute ramp changes over the range of 15% to 100% of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system.

The diversion of a portion of the steam in the cold reheat piping to the new reboiler has the potential to cause the turbine control system to produce changes in steam flow from the steam generators to the main turbine. Analyses (see Reference 3, Section 5.1.2.5) have determined that the maximum proposed diversion of a portion of the cold reheat steam to the HTEF at full power would result in an approximately 2.99% decrease in the normal full-power cold reheat flow from the high-pressure (HP) turbine to the moisture separator reheaters (MSRs) and an approximately 3.18% decrease in the hot reheat flow out of the MSRs. The change in hot reheat steam flow to the low-pressure (LP) turbines will result in a small reduction (approximately 1.81%) in the megawatt electric output of the main generator.

A change from the maximum to the minimum (or from the minimum to the maximum) cold reheat steam flow to the HTEF may produce a slight plant transient, but the transient conditions are well within the capabilities of the turbine control system and the other plant control systems. In addition, existing plant procedures already contain guidance to minimize the risk from unexpected balance-of-plant transients near full-power conditions. Since these small transient conditions are within the normal operation of the plant control systems, they do not constitute an "accident previously evaluated in the FSAR," as that phrase refers to abnormal operational transients and postulated design basis accidents that are analyzed to demonstrate that the facility can be operated without undue risk to the health and safety of the public (see NEI 96-07, Section 3.2).

Therefore, a sudden change in steam flow to the HTEF does not result in more than a minimal increase in the frequency of occurrence of either an excess steam flow event or a loss of load event.

Loss of External Electrical Load

The diversion of electrical power from the high-voltage side of the main transformer introduces the potential for a transient condition at the main generator or switchyard. Such a transient could potentially create generator instability that could lead to a Loss of External Electrical Load occurrence.

Sensitivity analysis performed and documented in Reference 3 shows that the HTEF load can be increased up to the maximum output power rating of the generator without causing the generator to become unstable following a trip of the high-voltage transmission line feeding the HTEF, either with or without a fault. During a trip of the line (for either faulted or un-faulted case) for the 500-MW_{nom} design, the generator exhibits a temporary increase in mechanical speed, followed by damped oscillations.

These speed changes would be within the normal operational capabilities of the turbine control system and the other plant control systems (i.e., turbine-generator does not trip when controls function as designed) and thus do not constitute an "accident previously evaluated in the FSAR." Therefore, a sudden change in the electric power to the HTEF does not result in more than a minimal increase in the frequency of occurrence of a loss of load event.

The installation of new electrical devices at the high-voltage side of the main transformer to provide electrical power for the HTEF introduces the potential for an electrical fault between the connection and the new high-voltage breaker or for spurious actuation of the associated protective devices to cause a trip of the main generator. Such a fault could potentially lead to a Loss of External Electrical Load occurrence.

However, existing high-voltage electrical equipment is designed and maintained to the codes and standards and practices appropriate for this application, such that these types of spurious faults or failures are rare. Because the new electrical equipment will likewise be designed and maintained to the appropriate codes and standards and practices, the likelihood of a fault or failure in the new equipment remains very low. Therefore, there is no more than a minimal increase in the frequency of occurrence of a generator fault being induced with the addition of the new electrical equipment to support the HTEF.

Loss of Offsite Power

Protective relaying for the new high-voltage electrical equipment from the main transformer to the HTEF would prevent a failure or fault in that equipment from affecting the offsite power supply, which is connected to the switchyard at different locations. Because the new relaying will be designed to the same codes and standards as similar existing relaying, the likelihood of a failure in the new relaying remains very low. Therefore, there is no more than a minimal increase in the frequency of occurrence of a loss of offsite power event.

This conclusion is supported by the PRA results in Reference 1.

High-Energy Line Break

The routing of steam in the new piping from each cold reheat pipe to the new reboilers introduces the potential for a failure in the new piping or in associated new equipment that could result in a high-energy line break in the plant. Steam piping and components in nuclear power plants are designed, constructed, and operated to ANSI/ASME codes and standards for piping, such that failures in steam piping and components that lead to high-energy line breaks are exceedingly rare. The new piping and components represent a small addition to the amount of high-energy piping and the number of high-energy components already in the plant. Because the new piping and components will be designed to the same codes and standards as the existing cold reheat piping, the likelihood of a failure in the new piping and components remains very low. Therefore, there is no more than a minimal increase in the frequency of occurrence of a high-energy line break.

This conclusion is supported by the PRA results in Reference 1.

Flooding

The new piping and components from the new reboilers to the condenser and the new demineralized water tanks and associated piping and components introduce the potential for a failure that could result in flooding. The new piping and components represent a small addition to the existing piping and components, which have the potential to initiate a flooding event. Since the new piping, tank, and components will be designed to the same codes and standards as similar existing piping, tanks, and components, the likelihood of a failure in the new piping and components remains very low. Therefore, there is no more than a minimal increase in the frequency of occurrence of flooding.

Turbine Trip

The diversion of a portion of the steam in the cold reheat piping to the new reboiler has the potential to cause the turbine control system to produce changes in steam flow from the steam generators to the main turbine. This interaction is described in "Excessive Increase in Steam Flow". As stated above, a change from the maximum to the minimum (or from the minimum to the maximum) cold reheat steam flow to the HTEF may produce a slight plant transient, but the transient conditions are well within the capabilities of the turbine control system and the other plant control systems. Therefore, there is no more than a minimal increase in the frequency of occurrence of a turbine trip.

Hydrogen Hazards

Hydrogen hazards are not evaluated in the UFSAR. The potential for a catastrophic fire or explosion at the HTEF to initiate an accident previously evaluated in the UFSAR is addressed in the response to Question 5.

Question 2: Does the proposed activity result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR?

The new steam, condensate, demineralized water, and high-voltage electrical equipment are not themselves SSCs that perform a function that is important to safety. The performance of the new equipment under normal operation or under anticipated operational occurrences has minor effects on plant systems (as discussed in the response to Question 1) and does not result in the malfunction of SSCs important to safety. (Note that significant transient events or conditions are tracked under an existing station program to ensure that fatigue limits on SSCs are maintained.)

The installation of new steam, condensate, and demineralized water piping and components and the installation of new high-voltage electrical equipment introduces the potential for a catastrophic failure in any of these to affect SSCs important to safety. As discussed in the response to Question 1, the codes, standards, and practices used in the design, construction, and operation of such items will provide sufficient assurance that the likelihood of a catastrophic failure is very low. In addition, the routing of the piping and location of new components minimizes or eliminates the potential for a catastrophic failure to cause a malfunction of an SSC important to safety. Therefore, there is no more than a minimal increase in the likelihood of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The proposed activity includes the installation of several new instrumentation and control devices at the main control board. Operation or mis-operation of these instruments and controls cannot cause a malfunction of an SSC important to safety, and the new devices will not interfere with the operation of existing equipment in the main control room. This is assured because the new simplified steam HTEF controls are not integrated into the nuclear power plant control systems and thus any failures of new controls are not capable of impacting existing systems with ties to SSC functional controls. Select information from the new equipment and controls is available in the main control room; the routing of this information will not adversely affect existing information systems in the main control room.

The potential for a catastrophic fire or explosion at the HTEF to cause malfunctions of SSCs important to safety is addressed in the response to Question 5.

Question 3: Does the proposed activity result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR?

As discussed in the response to Question 1, the performance of the new equipment under normal operation or under anticipated operational occurrences has minor effects on plant systems and does not result in any accident that has radiological consequences.

The installation of new steam piping and components introduces the potential for a failure in any of these that could result in a high-energy line break in the plant. The existing plant systems include steam piping and associated components whose failure could have radiological consequences. The bounding analysis for such events is the analysis of the radiological consequences of the rupture of a main steam line. That analysis assumes that primary to secondary leakage is released directly from the faulted steam generator to the environment and also through the power-operated relief valves of the unfaulted steam generators. The analysis remains bounding for the case of a rupture in the much smaller steam lines to be installed under the proposed activity.

The installation of the new equipment does not affect the response of SSCs credited for accident mitigation. The installation of the new equipment, including new high-voltage electrical equipment and associated protective relaying to provide power to the HTEF, does not affect the post-accident response of the main generator or plant electrical systems.

Control failures and the potential for new equipment response impacts are addressed in the response to Questions 1 and 2.

Therefore, the proposed activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

Question 4: Does the proposed activity result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR?

The installation of new steam, condensate, and demineralized water piping and components and the installation of new high-voltage electrical equipment does not introduce the possibility of a change in the consequences of a malfunction because the new equipment is not an initiator of any new malfunctions of SSCs that could lead to or mitigate radiological consequences and no new failure modes of such SSCs are introduced.

Control failures and the potential for new equipment response impacts are addressed in the response to Questions 1 and 2.

Therefore, the proposed activity does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Question 5: Does the proposed activity create a possibility for an accident of a different type than any previously evaluated in the UFSAR?

General Design Criterion 4 in Appendix A to 10 CFR Part 50 includes a requirement that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions and dynamic effects associated with postulated accidents that may result from events and conditions outside the nuclear power unit. The HTEF, which will be producing hydrogen nearby, introduces the potential for an explosion or fire at a nearby facility to affect SSCs onsite.

Evaluations performed for the siting of a generic HTEF assumed that the HTEF electrolyzers would be located at least 500 m from SSCs important to safety [Reference 1]. This minimum distance represents the point at which the overpressure effect from the maximum credible accident at the HTEF would fall below 1 psi. At this distance, a deflagration event is expected to have no consequence to the nuclear plant.

The UFSAR for the reference plant used in developing this report discusses the potential for an explosion at a nearby facility or on a nearby transportation route. An evaluation compared the probability of an explosion to the acceptance criteria of Regulatory Guide 1.91 ("Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants"), Rev. 1 [Reference 2] and concluded that an explosion on a nearby route was not a credible event.

Revisions 1 (February 1978), 2 (April 2013), and 3 (November 2021) of the current version of Regulatory Guide 1.91 each provide: (1) a method for determining the distance from critical plant structures to the location of the explosion, a distance beyond which any explosion is not likely to have an adverse effect on plant operation or prevent a safe shutdown, and (2) methods for determining whether the frequency of occurrence of an explosion is sufficiently low. Although there are differences between these various versions of the Regulatory Guide, the basic methodology for determining the distance beyond which any explosion is not likely to have an adverse effect on plant operation or prevent a safe shutdown has not changed.

An evaluation of the explosion hazard from the proposed HTEF was not performed. However, based on the amount of hydrogen that would be released from a failure at the HTEF, it is expected that such an evaluation would show that the relevant Regulatory Guide 1.91 criteria for the distance from the HTEF to SSCs important to safety is met. Therefore, the presence of the HTEF would not create a possibility for an accident of a different type than any previously evaluated in the UFSAR. This is supported by the conclusions of the explosion hazard analysis performed in Reference 1. Reference 1 analyzed an HTEF with hydrogen volumes bounding the proposed pre-conceptual design; this analysis determined that a 500 m standoff distance to the most fragile nuclear SSCs, the switchyard transmission towers, is viable.

Operator errors involving the new operator controls for steam and electric power to the HTEF could initiate, at most, small transients in balance-of-plant systems. As discussed in the response to Question 1, changes that might result in these small transients are within the normal operational capabilities of the turbine control system and the other plant control systems. Such transients do not reach the threshold of an "accident previously evaluated in the FSAR." In addition, such minor transients do not constitute an accident of a different type.

Control failures and the potential for new equipment response impacts are addressed in the response to Question 1 and Question 2.

Therefore, the proposed activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR.

Question 6: Does the proposed activity create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR?

As discussed in the response to Question 2, the installation of new steam, condensate, and demineralized water piping and components and the installation of new high-voltage electrical equipment introduce the potential for a catastrophic failure in any of these to affect SSCs important to safety. The new piping and components will be located such that there is no potential for a catastrophic failure to cause a malfunction of an SSC important to safety with a different result (e.g., failure of redundant SSCs important to safety) than any previously evaluated in the UFSAR.

As discussed in the response to Question 5, it is expected that an evaluation of the explosion hazard from the proposed HTEF would show that the relevant Regulatory Guide 1.91 criteria for the distance from the HTEF to SSCs important to safety is met. Therefore, the presence of the HTEF would not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

Question 7: Does the proposed activity result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered?

The installation of new steam, condensate, and demineralized water piping and components and the installation of new high-voltage electrical equipment introduces the potential for failures of these new SSCs to produce small changes in the flow of steam, condensate, and demineralized water, and also changes to the electrical power system. Such changes are not associated with any fission product barrier. Therefore, the proposed activity does not result in a design basis for a fission product barrier as described in the UFSAR being exceeded or altered.

Question 8: Does the proposed activity result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses?

The methods of evaluation used in determining the effects of a transient in the cold reheat or other extraction steam systems are not described in the UFSAR. The methods of evaluation used in determining the effects of a transient in the plant electrical systems are not described in the UFSAR for the reference plant.

Relatively small quantities of hydrogen are currently stored at the site for use in the main generator and in the chemical and volume control system. No methods of evaluation for determining the impact of a hydrogen fire or explosion on site are described in the UFSAR.

The proposed activity involves the production and transportation of larger quantities of hydrogen near the site. As discussed in the response to Question 5, it is expected that an evaluation of the explosion hazard from the proposed HTEF would show that the relevant Regulatory Guide 1.91 criteria for the distance from the HTEF to SSCs important to safety is met. Because such an evaluation would use the same method of evaluation used in Regulatory Guide 1.91, the method of evaluation would be one approved by the NRC for the intended application.

Therefore, the proposed activity does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

4. CONCLUSION

Generic 10 CFR 50.59 evaluation of the pre-conceptual design described in Reference 3 concludes that a 500 MW_{nom} high-temperature electrolysis facility addition to a nuclear power plant can be performed under the 10 CFR 50.59 process for many plants within the existing U.S. nuclear fleet. This was determined based on the limited impact of the hydrogen production facility on the reference plant mechanical, electrical, and controls systems.

This generic evaluation assumed the presence of an existing analysis for explosive hazards in the vicinity (i.e., explosions at nearby facilities or on nearby transportation routes) of the nuclear plant. Hazard analysis would need to be performed to assess whether there is (1) the possibility for an accident of a different type than any previously evaluated in the UFSAR and (2) the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR. Nuclear plants that do not have an existing hazard analysis meeting these requirements may need to pursue a license amendment request.

This evaluation is specific to the facility design and reference plant documented in Reference 3, and cannot be used as a template for other nuclear power plants. Consultation with the nuclear steam supply system and turbine generator vendors should be performed as needed to validate plant responses for the various failure scenarios considered in this evaluation. Other plants may have more detailed discussion of methods of evaluation, additional accidents or transients considered in their UFSAR, and unique design features which would need to be addressed in the project-specific evaluation responses.

5. REFERENCES

1. INL/EXT-20-60104, Rev. 1, *"Probabilistic Risk Assessment of a Light Water Reactor Coupled with a High-Temperature Electrolysis Hydrogen Production Plant,"* Vedros/Christian/Rabiti, November 2022.

2. RG 1.91, Rev. 1, "*Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes Near Nuclear Power Plants,"* U.S. Nuclear Regulatory Commission, February 1978.

3. SL-016181, Rev. 1, "Nuclear Power Plant Pre-Conceptual Design Support for Large-Scale Hydrogen Production Facility," Sargent & Lundy, November 2022.

Appendix B includes this latest generic 10 CFR 50.59 evaluation in support of implementing a 500 MW_{nom} modification as a proposed change to an assumed 1200 MWe GPWR. This report [9] concluded that many plants could add a 500 MW_{nom} HTEF using a 10 CFR 50.59 evaluation if two conditions are met: (1) the impact of the hydrogen production facility on the reference plant mechanical and electrical systems is limited, which has been shown in the generic analysis; and (2) an analysis of explosive hazards in the vicinity of the plant per [4 and 5] is part of the current licensing basis for the reference plant.

Subcommittee-Specific Considerations

Appendix C includes noteworthy H3RG working subcommittee excerpts regarding design assumptions, references, and regulatory considerations that should be considered in the plant-specific FEED studies. In addition to evaluation areas described in Appendix C, H3RG Regulatory Subcommittee assessments also identified siting impacts that could require a TS revision including:

1. The specific location of the HTE equipment may be important to prevent a challenge to existing plant equipment and features necessary to mitigate a DBA or transient. This concern is relevant to Criterion 3 of 10 CFR 50.36.
2. A recent industry case in which a map of the owner-controlled area features was included in the plant TS. The NRC recently cited this licensee for revising this map to reflect an alternate use of a part of the owner-controlled area without prior NRC approval. This ongoing discussion is related to 10 CFR 50.83 requirements regarding partial release of the NPP site for unrestricted or alternate uses. This type of TS issue and others are expected to be highly plant-specific based on the content of individual Technical Specifications. This may include site layout-related information that might be contained in the Administrative Section of TS. Additionally, introducing nuclear integrated HTE on a NPP site may be viewed as effectively changing site use intent as contained or implied in the TS and could require certain notification and reporting in addition to reporting of 50.71(e) UFSAR and 50.59 updates to the NRC.
3. As described under Section 2.3, depending on whether the original licensing basis was evaluated and approved in consideration of RG 1.91, a license amendment may be required to implement a new hydrogen detonation analysis methodology. This is yet to be fully determined and is one of the topics being explored as described under Section 2.4 Summary Conclusions and Next Actions.
4. It is also expected that additional site-specific licensing-related evaluations will be required in the following areas:

- Security Plan – 10 CFR 50.54(p)
- Emergency Plan – 10 CFR 50.54(q)
- QA Topical Report – 10 CFR 50.54(a)
- ISFSI considerations – 10 CFR 72.48.
- Control Room Habitability – 10 CFR 50 App. A

There are currently no plans for additional H3RG work to address a more comprehensive list of generic site-specific considerations since site-specific reviews will naturally emerge and need to be addressed for individual near-term iFOA award plants performing FEED studies. Operating experience from these evaluations may be considered in the future for applicability in developing generic licensing approaches.

2.2.2 Hydrogen Detonation Analysis Methodology Considerations

The H3RG identified the need to differentiate between plants originally licensed with consideration of potential explosive hazards and those that were not. This latter group of plants may not pass 10 CRF 50.59 evaluation question 5 regarding the possibility for an accident of a different type than any previously evaluated in the FSAR due to hydrogen detonation events. This was specifically considered most likely in cases where original licensing did not consider evaluation under Regulatory Guide 1.91 [5] related to explosions near nuclear power plants. These plants would likely need an LAR that addresses the new explosive hazard presented by the HTEF.

The most current 10 CFR 50.59 evaluation developed for HTEF's up to 500 MW_{nom} [9] assumed that the amount of hydrogen that would likely be released from a failure at the HTEF to SSCs important to safety would be within the earlier bounding 1 km stand-off distance risk analysis conclusions. Based on the FMEA and detonation evaluation work performed under [4 & 7] using HyRAM+, this earlier conclusion is *technically supported* for the generic PWR and HTEF layout [6]. This is discussed in more detail in the recently issued PRA [14].

Although favorable to the goal of this research, these more realistic HyRAM+-based results are not consistent with the RG 1.91 "equivalent TNT" methodology [4]. It is expected that plants seeking to evaluate postulated detonation effects on NPP SSCs from a proposed HTEF would determine bounding scenarios for hydrogen vapor clouds, jet leakage, and mass release based on failure modes and effects analysis and details of the site HTEF layout (like that described under [6, 7 and 14]) with spatial consideration to NPP SSCs as graphically represented in Figure 9.

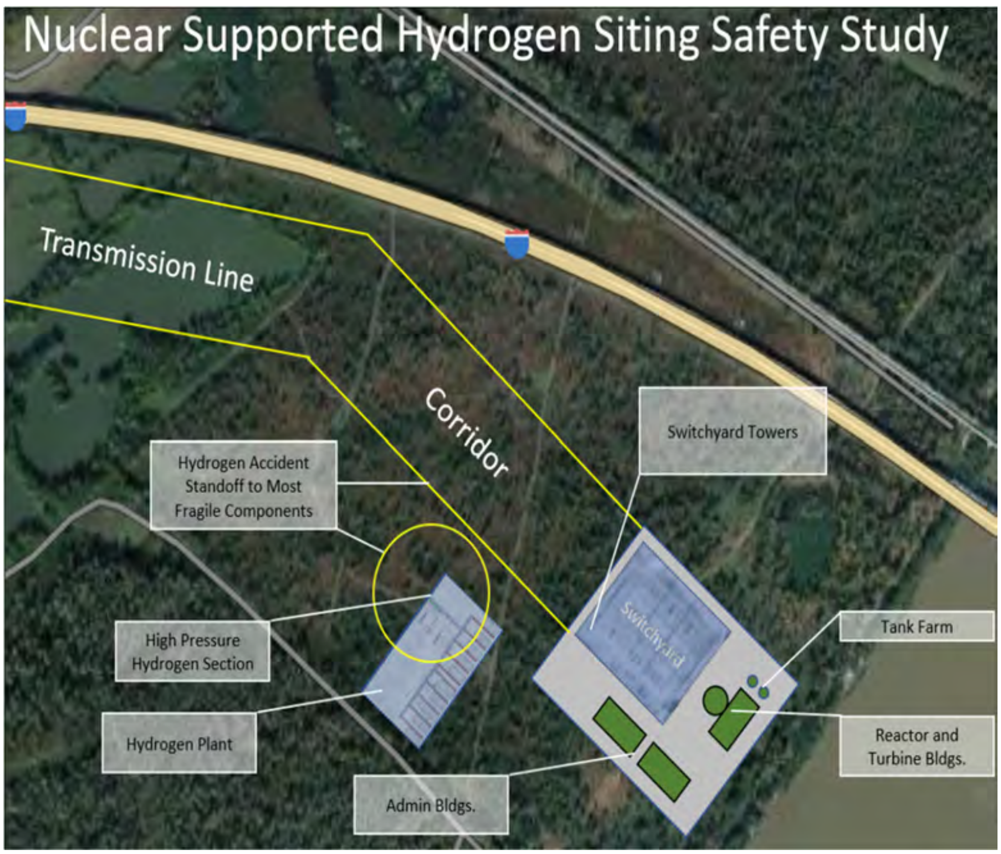

Figure 9. Generic Hydrogen Facility siting map.

If calculated HTEF standoff distances were to support the proposed siting layout and safe distances to SSCs, it is likely that the RG 1.91 methodology would be employed directly as a basis for a focused LAR.

However, since RG 1.91 specifies a standoff distance for a 1 psi threshold, it does not protect against the increase in the LOOP initiating event frequency caused by the damage of the transmission tower (lines) at greater than 0.16 psi overpressures. In addition, if site congestion or other considerations preclude the planned HTEF location using the RG 1.91 methodology, the HyRAM+ Bauwens methodology could be considered as alternate approach for evaluation of explosions postulated to occur near nuclear power sites. As shown in [6] the HyRAM+ Bauwens methodology yields more realistic standoff distances from HTEF to NPP SSC's. Both the discussion of RG 1.91distancing applicability to protecting from the LOOP initiating event increase and the HyRAM+ Bauwens approach would require NRC interaction and acceptance. Part of the regulatory discussion around this would be related to the comparative value of the Bauwens methodology as a rigorous *hydrogen-specific* validated analysis compared to 1970's era generic RG 1.91 analysis methodology. Additionally, depending on the number of plants seeking such approval under HyRAM methodology it may be preferrable to develop a topical report submittal support approach. This is one of the topics addressed under Section 2.4.1, Summary Conclusions and Next Actions.

2.3 Supporting Research Deliverables

2.3.1 Research Interconnections

Revision 2 provides a description of the major supporting nuclear-integrated hydrogen research deliverables (with a status of previously issued or soon-to-be issued) as a guide for industry licensees and architect engineers who may be developing plant-specific design and regulatory approaches. These research deliverables provide valuable design and licensing considerations for nuclear utilities and architect engineers who will be developing plant-specific medium and large-scale demonstration or commercial designs. The major supporting research documents are graphically represented by year in Figure 10 and in more detail under Appendix D with a hyperlink to the LWRS report website, https://lwrs.inl.gov/SitePages/Reports.aspx.

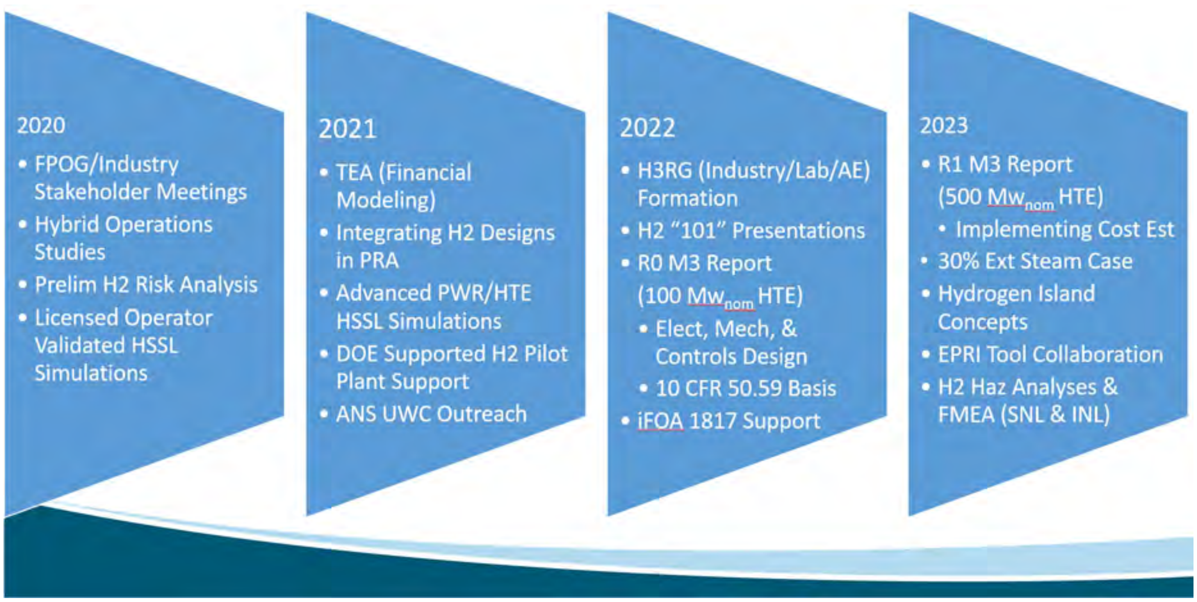

Figure 10. Progressive R&D support of nuclear integrated hydrogen (HTE).

2.4 Summary Conclusions and Next Actions

2.4.1 Summary Conclusions

Ongoing plant analytical integration efforts (including the use of HyRAM+) continue to provide favorable results in related areas such as plant integration response and associated risk analysis.

The 10 CFR 50.59 completed draft deliverable contained in Appendix A provides a framework to assess the eight criteria of 10 CFR 50.59 (c)(2). Reference [9] provides the continuing foundational confidence to develop conceptual 100 MW_{nom} to 500 MW_{nom} nuclear integrated hydrogen HTE designs at a standard 1200 MW generic PWR NPP. The intent of these design and associated regulatory assessments was to determine compatibility in support of the next generation of MW-level nuclear integrated HTE demonstration/commercial projects. From a high-level summary perspective, the following research findings apply:

- Plants desiring to add *nuclear integrated hydrogen by HTE* up to 500 MW_{nom} plant size should be able to justify the plant modification without formal license amendments (LARs) under the 10 CFR 50.59 licensee evaluation process if their original licensing basis included consideration of the effects of explosion of hazardous materials (including hydrogen gas) at or near the NPP.
    1. The term "nuclear integrated hydrogen by HTE" is only intended to describe the connected steam extraction and electrical supply from the NPP to the hydrogen facility boundary as shown in Figure 5.
        2. Although Reference [14] also simplistically evaluated a 1000 MW_{nom} case from a hazard standpoint, it has yet to be supported with the same design and 10 CFR 50.59 evaluation rigor as provided for the 100–500 MW_{nom} cases. It should therefore only be considered with that current limitation in mind.
- Plants without specific mention of explosion hazards (e.g., RG 1.91 [4]) in their licensing basis will need to perform a plant-specific detonation hazard analysis to show there an assessment of no-adverse-impacts to SSC's. Plants that need closer standoff distances than those calculated by the RG 1.91 TNT equivalent methodology could potentially seek licensing approval under an alternate basis. This report revision 2 and [14] present such a design and risk case for HyRAM+ detonation methodology. Section 2.5.2 addresses next steps to seek approval bases with the NRC on potential hydrogen detonation evaluation options including:
    1. Direct use of the RG 1.91 [4] methodology to determine hydrogen plant standoff distances such that explosive external dynamic peak overpressure loading effects calculated by equivalent weight of explosive (TNT) are enveloped by the regional design basis tornado wind load as described in Reg Guide 1.76 Design Basis Tornado for Nuclear Power Plants [5]. This assumes an original licensing basis that included postulated explosions near the NPP.
    2. Alternate use of HyRAM+ methodology to directly calculate safe standoff distances based on rigorous siting and FMEA considerations. The H3RG recommended HyRAM+ analyses necessary to generically demonstrate the feasibility of this option (as described under Sections 2.1.2 and 2.2.3). This analysis has been completed as a 2023 R&D deliverable for the 100 and 500 MWnom nuclear integrated hydrogen HTE designs on the generic 1200 MW PWR.
- As described under Section 2.5.2, future actions are planned for NRC discussions to explore approval of HyRAM+ (or other such hydrogen detonation modeling tools) to meet the detailed standoff distance approach of RG 1.91 where plant specific siting considerations cannot support direct use of the RG 1.91 equivalent TNT methodology, or the use of the RG is otherwise unacceptable. The goal is identifying a path for industry use of this alternate approach in the 2024-2025 timeframe.
    1. As part of Revision 2 to this report, Section 2.1.3 introduces interest in furthering industry/NRC dialog to potentially include the use of abbreviated process for approval of alternate detonation risk evaluations (including through generic licensing vehicles like topical reports).
- Follow-on review of IP gas transmission line lessons learned represents an opportunity to create one element of an NRC discussion framework to comparatively discuss Indian Point regulatory approval and laboratory research approaches and how laboratory research deliverables now reaching intended maturity address such concerns. This is captured for further action under Section 2.4.2.

- Plant unique licensing considerations will need to be evaluated and potentially submitted for regulatory approval for:
  - Emergency Plans
  - Security Plans
  - Independent Spent Fuel Storage Facilities
  - QA Plans
  - Control Room Habitability
  - Technical Specification language that may conflict with co-locating a nuclear integrated hydrogen facility in the Owner Control Area
  - Compliance with regulations outside the purview of 10 CFR 50.59 (i.e., other federal and state regulations and local ordinances, as applicable)

The latest 100 MW_{nom} to 500 MW_{nom} nuclear integrated hydrogen HTE design and generic 50.59 evaluation represents a reasonable starting point for plants considering the proposed addition of a hydrogen production facility. It is recognized that the findings of this generic 10 CFR 50.59 evaluation are based on pairing with the GPWR design. For plants for which the GPWR modeling does not represent the site-specific licensing requirements, license amendments or additional research will be needed to mitigate these challenges. Thus, this emerging 10 CFR 50.59 template is expected to be useful to industry with varying degrees of customization. Plant-specific FEED studies are the best way to identify the final technical and regulatory level-of-fit for nuclear plants desiring to adopt hydrogen as an alternate energy stream.

2.4.2 Next Actions

The following near-term nuclear integrated HTE research actions are condemplated:

1. Continued H3RG Regulatory Strategy/Oversight subcommittee participation with a focus on development of regulatory approval strategies for HyRAM+ or other similar hydrogen-specific detonation analysis methodologies. Such approaches would explore streamlined regulatory approval paths for utilities not originally licensed to RG 1.91 methodology, those seeking closer stand-off distances between HTEF components and SSC’s due to site special constraints, and the appropriate use of RG 1.91 distancing for lower than 1.0 psi susceptible SSCs.
2. Perform a follow-on review of IP gas transmission line lessons learned to support one element of an NRC/laboratory discussion framework to comparatively discuss Indian Point regulatory approval and laboratory research approaches and how laboratory research deliverables now reaching intended maturity address such concerns.
3. Establish periodic briefings between DOE (INL) and NRC (Research and Nuclear Reactor Regulation Offices) under Laboratory Memorandum of Understanding on interaction areas related to regulatory approval of nuclear integrated hydrogen by HTE including:

- Information sharing approaches for work in progress including LWRS FPOG Public Source R&D report website, etc.
- Present applicable nuclear integrated hydrogen research findings to date and solicit feedback. Include early discussions on comparative evaluation of Indian Point regulatory approval and laboratory research approaches and how laboratory research deliverables address or need to further address such concerns.
- Possible use of abbreviated licensing approval tools to adopt HyRAM+ (or other) hydrogen detonation evaluation methodology for utilities not licensed to RG 1.91 or for which use of RG 1.91 equivalent TNT methodology yields impractical NPP to HTEF standoff distances:
  – Possible DOE/industry sponsored topical report.

  - Graded regulatory approval approaches to reduce regulatory review time.
  - Risk Informed Process for Evaluation (RIPE).
- General guidance for individual utilities to address plant-specific licensing requirements, e.g., Security and Emergency Plan hydrogen plant siting impacts.
- Updates on DOE co-funded development awards:
  - Pilot projects and laboratory HTE equipment demonstrations
  - AE funded R&D supporting evaluations and studies
  - Advanced simulation research on cogeneration coupling of nuclear plant electrical and heat capabilities with non-electrical products
  - International IAEA collaborations
  - iFOA awards and laboratory support of Front-End Engineering and Design (FEED studies)
  - Updates on laboratory-led techno-economic evaluations and nuclear based hydrogen HUB development support progress
  - Integrated Energy Systems early support of advanced reactors
  - Progress on next-generation laboratory-led technical and regulatory research on non-electrical products "beyond hydrogen."

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Appendix A

10 CFR 50.59 Evaluation

SL-017337 R1 - "Nuclear Power Plant Pre-Conceptual 10 CFR 50.59 Evaluation for Large-Scale Hydrogen Production Facility"

Nuclear Power Plant Pre-Conceptual Licensing Support for Large-Scale 500-MW_{nom} Hydrogen Production Facility

Report SL-017513

Revision 1

May 13, 2023

Project No.: A14248.009

S&L Nuclear QA Program Applicable:

☐ Yes

☒ No

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LIMITATIONS OF USE

This design report is provided as a guide and feasibility assessment for successful implementation of the 10 CFR 50.59 evaluation process for coupling a large-scale hydrogen production facility with a commercial nuclear power plant. Evaluations within this report are provided for the reference nuclear power plant and 500-MW$_{nom}$ hydrogen production facility described in Reference 3. The results of these evaluations cannot be extrapolated for application to different sized plants or extraction quantities. Site-specific factors will affect the conclusions of plants.

Note that a separate 10 CFR 50.59 evaluation (S&L Report SL-017337 Rev. 1, November 2022) was previously performed for a pre-conceptual 100-MW$_{nom}$ hydrogen production facility [Reference 3] and some of the applicable information from that evaluation is replicated here for completeness, however each of the reports are separate.

The evaluations presented within this report are applicable to implementation at commercial nuclear power plants that utilize the pressurized water reactor design. Plants designed as boiling water reactors or other reactor designs different than the reference plant design discussed in Reference 3 are not in the scope of this report and would require additional considerations for successful implementation of a 10 CFR 50.59 evaluation for a large-scale hydrogen production facility.

ISSUE SUMMARY AND APPROVAL

This is to certify that this document has been prepared, reviewed, and approved in accordance with Sargent & Lundy's Standard Operating Procedure SOP-0405, which is based on ASQ/ANSI/ISO 9001:2015: Quality Management Systems–Requirements.

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EXECUTIVE SUMMARY

Nuclear power is a large-scale source of clean energy that can be used to advance national long-term goals of decarbonization. At the same time, nuclear power plants are facing economic pressures due to fluctuating electrical demand and low natural gas prices. These factors are driving strategic innovation in the commercial nuclear power industry to identify and develop alternative avenues of revenue. One current initiative involves coupling a nuclear power plant with a large-scale, high-temperature electrolysis hydrogen production facility.

In a previous report [Reference 3], the logistical and economic feasibility of a nuclear integrated hydrogen facility was investigated through a pre-conceptual design. The design implemented a high-temperature electrolysis facility with nominal power requirements of 500-MW_{nom}. The thermal and electrical design of integration between the nuclear and hydrogen plants was developed for a generic 1200-MW_e pressurized water reactor design, along with a cost estimate for the modification.

To aid utilities in pursuing their own 50.59 evaluations for future hydrogen production facilities, a draft 50.59 evaluation for the addition of a 500-MW_{nom} high-temperature electrolysis facility (HTEF) design to a reference plant has been developed and is provided in this report. Based on (1) the limited impact of the hydrogen production facility on the reference plant mechanical and electrical systems and (2) the existence of an analysis for explosive hazards in the vicinity of the plant as part of the current licensing basis for the reference plant, it is expected that a 500 MW_{nom} HTEF addition can be performed for many plants under the 10 CFR 50.59 process. However, plants that do not have an existing evaluation for explosive hazards in the vicinity of the plant as part of their licensing basis may need to pursue a license amendment request.

1. BACKGROUND

One of the focuses of the United States Department of Energy's (DOE) Light Water Reactor Sustainability (LWRS) program is to explore avenues that can extend the operation of the U.S. commercial nuclear power plant fleet. Within the LWRS program, the Flexible Plant Operation and Generation (FPOG) Pathway is working to diversify the revenue streams of light-water reactors (LWRs) through the exploration of nuclear power plant operation beyond supplying electrical power to the grid. Nuclear power has been identified as a source of large-scale, carbon-free "clean" steam, with thermal and electrical energy that can be utilized to realize national long-term goals of decarbonization.

U.S. nuclear plants are typically operated at full (100%) power to provide baseload electrical power to the national grid. In deregulated markets, these plants face economic pressures from fluctuating electrical demand and decreasing prices of wind and solar generation. Exploring alternative uses for the clean steam produced by nuclear power during these challenging times is critical to improving the viability of nuclear plant operation.

One area of research at the DOE's Idaho National Laboratory (INL) has been focusing on the use of clean steam produced by a nuclear power plant to support the production of hydrogen (H_2) through the emerging technology of high-temperature electrolysis (HTE). The combination of H_2 production, storage, and distribution, through what are known as "H_2 hubs" in support of the transportation, agricultural, and industrial sectors, has been identified as a strategic avenue to support overall decarbonization in the United States. Electrolysis is the process through which water is decomposed into its oxygen and hydrogen gases via the application of an electrical potential. Research in the field has shown electrolysis to be more efficient at elevated temperatures. The process of HTE leverages this advantage using high-temperature steam as the water source for the reaction. The steam is broken down using rectified direct-current (dc) power within a solid-oxide electrolyzer cell (SOEC) to produce H_2 that can then be compressed, liquified, stored, etc., depending on the intended application.

The Hydrogen Regulatory Research Review Group (H3RG), which is made up of representatives from nuclear utilities and national laboratories, along with consulting design engineering representatives from Sargent & Lundy (S&L), is supporting the LWRS FPOG Pathway. The H3RG is identifying licensing considerations associated with coupling a large-scale H_2 production facility to a commercial nuclear power plant. These considerations are discussed and evaluated within the H3RG subcommittees to determine whether it is feasible for a utility to perform this modification under the 10 CFR 50.59 process or whether a license amendment request (LAR) may be required for implementation.

1.1. Pre-Conceptual Hydrogen Production Facility Design

The proposed pre-conceptual study [Reference 3] establishes a high-temperature electrolysis facility (HTEF) with a nominal power requirement of 500 MW_{nom}. The parameters describing this facility are shown in Table 1 below.

Table 1. 500-MWnom Hydrogen Production Facility Pre-Conceptual Design Parameters

Parameter	Unit	Quantity
Hydrogen Production Capacity[1]	U.S. tons/day	300
H_2 Plant Electric Load	MW_e	500
Total Electrical Power Requirements	MVA	600
H_2 Plant Thermal Load	MW_t	100
Total Thermal Power Requirements	MW_t	105

[1] Conservative production capacities are shown based on 2022 values. Technology improvements are expected to increase production efficiencies over time.

1.2. Reference Nuclear Power Plant

Both thermal and electrical power are required for operation; this power is supplied by a nearby nuclear power plant. The nuclear reactor model and plant size can have significant effects on the integration with a hydrogen facility. Westinghouse 4-loop pressurized water reactors (PWRs) are the most prevalent reactor design in the United States, making up approximately one-third of the operational nuclear fleet; therefore, this design was selected as the reactor of choice for the reference plant developed. The generation capacity of this design is approximately 1200 MW_e. Minimal siting restrictions were included in the development of the general site layout. The switchyard was located adjacent to the Protected Area.

1.3. Siting of the Hydrogen Production Facility

The pre-conceptual design located the electrolyzers 500 meters (m) from the nuclear power plant's important-to-safety equipment, including the switchyard and transmission towers, based on the conclusions of a generic probabilistic risk assessment (PRA) performed for a similar conceptual design [Reference 1]. Hydrogen produced at the HTEF would be transported to a storage facility located at least 5 kilometers (km) from the nuclear plant.

1.4. Plant Interfacing

New piping connected to the exhaust of the main turbine (cold reheat lines) will provide steam to the plant secondary side of new heat exchangers (steam reboilers), which will then supply steam to the HTEF via new tertiary loops (one loop per reboiler). The condensate that forms on the plant secondary side of the reboilers will be returned to the main condenser. Demineralized water in each tertiary loop will be provided from their associated new storage tank and pump to be located onsite (in the Protected Area). Demineralized water will be supplied by a skid inside the HTEF boundary. Electric power for equipment controls will be provided by a new load center powered from the turbine-building portion of the station auxiliary power system.

Electric power to the HTEF will be provided from a new connection at the high-voltage side of the generator step-up transformer via two new manually operated disconnect switches, associated high-voltage electrical metering and relaying, a high-voltage circuit breaker, a high-voltage transmission line and associated towers that run to the HTEF facility. The end of the line at the HTEF facility will have two step-down transformers, which will step the power down from 345 kV to 34.5 kV. Each transformer will have one 345-kV circuit breaker and two 345-kV disconnect switches. Two outdoor 34.5-kV buses with nine 34.5-kV breakers, each connected to a step-down transformer, will step the power down from 34.5-kV to 13.8-kV switchgears.

The new equipment and the offsite HTEF are independent of existing onsite systems that provide hydrogen for the main generator and the chemical and volume control system. The new equipment to be located onsite and all the existing station equipment directly affected by the proposed activity are non-safety-related. The HTEF will include minimal hydrogen storage capability; hydrogen will be piped to an offsite facility for storage.

A general site layout of the pre-conceptual design considered in this report is illustrated below in Figure 1.

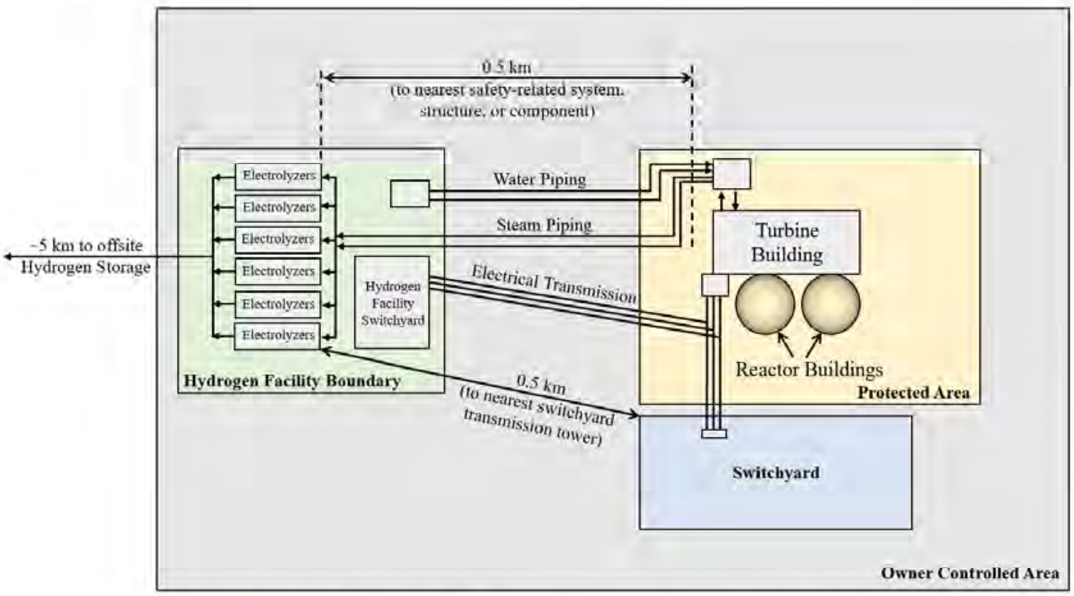


Figure 1. General Site Layout for Pre-Conceptual 500-MW$_{nom}$ Hydrogen Facility Design

2. INTRODUCTION

Section 50.59, “Changes, tests, and experiments.” of Title 10 of the Code of Federal Regulations (10 CFR) establishes the conditions under which licensees may make changes to the facility or procedures and conduct tests or experiments without prior United States Nuclear Regulatory Commission (NRC) approval.

The licensee follows a detailed process for any change to confirm that eight criteria provided in 10 CFR 50.59(c)(2) are met. If any of the referenced criteria are not met, then the 10 CFR 50.59 process cannot be used to implement the modification and a LAR must be submitted to the NRC for review and approval in accordance with 10 CFR 50.90, “Application for amendment of license, construction permit, or early site permit.” In addition, if the proposed change would require a change to the Technical Specifications, a LAR must be submitted. Finally, the provisions of 10 CFR 50.59(c)(2) do not apply to changes to the facility or procedures when the applicable regulations establish more specific criteria for accomplishing such changes.

The 10 CFR 50.59 Evaluation provided below is intended to help licensees who are considering implementing modifications similar to that described in Reference 3. Note that additional licensing-related considerations will also be required, including but not limited to the following:

- Security Plan - 10 CFR 50.54(p)
- Emergency Plan - 10 CFR 50.54(q)
- QA Topical Report - 10 CFR 50.54(a)
- ISFSI Considerations - 10 CFR 72.48
- Antitrust Considerations, listed as an Appendix to the Operating License
- Compliance with regulations outside the purview of 10 CFR 50.59 (i.e., other federal and state regulations and local ordinances, as applicable)

If any part of the HTEF were to be located within the owner-controlled area or the Protected Area, demonstration of compliance with additional regulations may be required (e.g., the radiation protection requirements of 10 CFR 20 for a restricted area); however, the HTEF would still be considered a nearby facility, as discussed in the 50.59 evaluation provided below.

A review of the station licensing documentation will need to be performed on a site-specific basis as part of the design change process, including the 50.59 process. It is expected that the plant Updated Final Safety Analysis Report (UFSAR) will need to be revised to describe the hydrogen plant and associated plant components.

Note that the 50.59 evaluation provided below is specific to the facility design and reference plant (inclusive of the analyses and related assumptions) documented in Reference 3, and cannot be used as a template for other nuclear power plants. It is assumed that the reference plant has an existing hazard evaluation (for initial plant licensing) based on Regulatory Guide 1.91, Rev. 1, and that the UFSAR does not include a description for certain methods of evaluation, as discussed in the evaluation responses. Other plants may have more detailed discussion of methods of evaluation or additional accidents or transients considered in their UFSAR that would need to be addressed in the evaluation responses. Note that for plants without an existing evaluation for explosive hazards in the vicinity (i.e., explosions at nearby facilities or on nearby transportation routes), a LAR may be required. Finally, as part of the process to develop the detailed design for a specific site, consultation with the nuclear steam supply system (NSSS) / turbine generator (TG) vendor should be performed as needed to validate plant responses for the various failure scenarios considered.

3. 10 CFR 50.59 EVALUATION FOR 500 MW_{NOM} DESIGN

The following responses are provided for the modification described within this report.

Question 1: ***Does the proposed activity result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR?***

The diversion of steam from the cold reheat piping to the new reboiler introduces the potential for a transient condition in the steam systems. The diversion of electrical power from the high-voltage side of the main transformer introduces the potential for a transient condition at the main generator or switchyard. These have the potential to affect the frequency of occurrence of an excess steam flow event or a loss of load event.

The routing of steam in new piping from the cold reheat piping to the new reboiler introduces the potential for a failure in the new piping or associated components. This has the potential to affect the frequency of occurrence of a high-energy line break. The routing of condensate in new piping from the new reboiler to the main condenser and the routing of demineralized water from offsite to a new storage tank and to the reboiler introduce the potential for a failure in the new piping, tank, and associated components which would affect the frequency of occurrence of a flooding event. The installation of new electrical devices at the high-voltage side of the main transformer introduces the potential for an electrical fault or spurious actuation of protective devices to affect the frequency of occurrence of a main generator trip.

Modifications to the existing plant configuration through the addition of equipment supporting the operation of the HTEF will introduce the potential for failures, transients and occurrences that are associated with accidents that are analyzed in the UFSAR. However, any risks associated with the integration of the new equipment are mitigated through two primary approaches.

First, plant modifications are designed and implemented using the applicable codes and standards in place for design, materials, and construction. As described in NEI 96-07, Rev. 1, Section 4.3.1, this approach provides assurance that potential conditions that may result from the modification (i.e., the proposed activity) will not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

Second, it is important to note that while there are process interfaces between the nuclear power plant and the HTEF (steam supply, condensate return and electrical power supply), there is no interface between control systems for each. The HTEF controls are completely separate from the nuclear power plant control systems. New controls in the control room are remote manual control stations that do not integrate with any other existing control system. These new controls provide simple capability to allow the nuclear power plant operators to take actions to isolate the HTEF connections as needed. Therefore, interactions between the nuclear power plant and HTEF are limited to the process connections and components (piping, valves, breakers, etc.) and excludes any interfaces between nuclear power plant and HTEF system controls.

Plant modifications for integrating the HTEF into the nuclear power plant have the potential to increase the frequency of occurrence of accidents previously evaluated in the UFSAR. The accidents that are of potential interest are:

- Excessive Increase in Secondary Steam Flow
- Loss of External Electrical Load
- Loss of Offsite Power
- High-Energy Line Break
- Flooding
- Turbine Trip

The potential impacts to the accidents previously evaluated are discussed below.

Excessive Increase in Secondary Steam Flow

The diversion of steam from the cold reheat piping to the new reboiler introduces the potential for a transient condition in the steam systems. This has the potential to affect the frequency of occurrence of an Excessive Increase in Secondary Steam Flow.

An excessive increase in secondary system steam flow (or excessive steam flow event) is defined as a rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand.

To accommodate routine operations at the nuclear plant, the design basis for normal operation of the plant control systems includes 10% step changes and 5% per minute ramp changes over the range of 15% to 100% of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system.

The diversion of a portion of the steam in the cold reheat piping to the new reboiler has the potential to cause the turbine control system to produce changes in steam flow from the steam generators to the main turbine. Analyses (see Reference 3, Section 5.1.2.5) have determined that the maximum proposed diversion of a portion of the cold reheat steam to the HTEF at full power would result in an approximately 2.99% decrease in the normal full-power cold reheat flow from the high-pressure (HP) turbine to the moisture separator reheaters (MSRs) and an approximately 3.18% decrease in the hot reheat flow out of the MSRs. The change in hot reheat steam flow to the low-pressure (LP) turbines will result in a small reduction (approximately 1.81%) in the megawatt electric output of the main generator.

A change from the maximum to the minimum (or from the minimum to the maximum) cold reheat steam flow to the HTEF may produce a slight plant transient, but the transient conditions are well within the capabilities of the turbine control system and the other plant control systems. In addition, existing plant procedures already contain guidance to minimize the risk from unexpected balance-of-plant transients near full-power conditions. Since these small transient conditions are within the normal operation of the plant control systems, they do not constitute an "accident previously evaluated in the FSAR," as that phrase refers to abnormal operational transients and postulated design basis accidents that are analyzed to demonstrate that the facility can be operated without undue risk to the health and safety of the public (see NEI 96-07, Section 3.2).

Therefore, a sudden change in steam flow to the HTEF does not result in more than a minimal increase in the frequency of occurrence of either an excess steam flow event or a loss of load event.

Loss of External Electrical Load

The diversion of electrical power from the high-voltage side of the main transformer introduces the potential for a transient condition at the main generator or switchyard. Such a transient could potentially create generator instability that could lead to a Loss of External Electrical Load occurrence.

Sensitivity analysis performed and documented in Reference 3 shows that the HTEF load can be increased up to the maximum output power rating of the generator without causing the generator to become unstable following a trip of the high-voltage transmission line feeding the HTEF, either with or without a fault. During a trip of the line (for either faulted or un-faulted case) for the 500-MW_{nom} design, the generator exhibits a temporary increase in mechanical speed, followed by damped oscillations.

These speed changes would be within the normal operational capabilities of the turbine control system and the other plant control systems (i.e., turbine-generator does not trip when controls function as designed) and thus do not constitute an "accident previously evaluated in the FSAR." Therefore, a sudden change in the electric power to the HTEF does not result in more than a minimal increase in the frequency of occurrence of a loss of load event.

The installation of new electrical devices at the high-voltage side of the main transformer to provide electrical power for the HTEF introduces the potential for an electrical fault between the connection and the new high-voltage breaker or for spurious actuation of the associated protective devices to cause a trip of the main generator. Such a fault could potentially lead to a Loss of External Electrical Load occurrence.

However, existing high-voltage electrical equipment is designed and maintained to the codes and standards and practices appropriate for this application, such that these types of spurious faults or failures are rare. Because the new electrical equipment will likewise be designed and maintained to the appropriate codes and standards and practices, the likelihood of a fault or failure in the new equipment remains very low. Therefore, there is no more than a minimal increase in the frequency of occurrence of a generator fault being induced with the addition of the new electrical equipment to support the HTEF.

Loss of Offsite Power

Protective relaying for the new high-voltage electrical equipment from the main transformer to the HTEF would prevent a failure or fault in that equipment from affecting the offsite power supply, which is connected to the switchyard at different locations. Because the new relaying will be designed to the same codes and standards as similar existing relaying, the likelihood of a failure in the new relaying remains very low. Therefore, there is no more than a minimal increase in the frequency of occurrence of a loss of offsite power event.

This conclusion is supported by the PRA results in Reference 1.

High-Energy Line Break

The routing of steam in the new piping from each cold reheat pipe to the new reboilers introduces the potential for a failure in the new piping or in associated new equipment that could result in a high-energy line break in the plant. Steam piping and components in nuclear power plants are designed, constructed, and operated to ANSI/ASME codes and standards for piping, such that failures in steam piping and components that lead to high-energy line breaks are exceedingly rare. The new piping and components represent a small addition to the amount of high-energy piping and the number of high-energy components already in the plant. Because the new piping and components will be designed to the same codes and standards as the existing cold reheat piping, the likelihood of a failure in the new piping and components remains very low. Therefore, there is no more than a minimal increase in the frequency of occurrence of a high-energy line break.

This conclusion is supported by the PRA results in Reference 1.

Flooding

The new piping and components from the new reboilers to the condenser and the new demineralized water tanks and associated piping and components introduce the potential for a failure that could result in flooding. The new piping and components represent a small addition to the existing piping and components, which have the potential to initiate a flooding event. Since the new piping, tank, and components will be designed to the same codes and standards as similar existing piping, tanks, and components, the likelihood of a failure in the new piping and components remains very low. Therefore, there is no more than a minimal increase in the frequency of occurrence of flooding.

Turbine Trip

The diversion of a portion of the steam in the cold reheat piping to the new reboiler has the potential to cause the turbine control system to produce changes in steam flow from the steam generators to the main turbine. This interaction is described in "Excessive Increase in Steam Flow". As stated above, a change from the maximum to the minimum (or from the minimum to the maximum) cold reheat steam flow to the HTEF may produce a slight plant transient, but the transient conditions are well within the capabilities of the turbine control system and the other plant control systems. Therefore, there is no more than a minimal increase in the frequency of occurrence of a turbine trip.

Hydrogen Hazards

Hydrogen hazards are not evaluated in the UFSAR. The potential for a catastrophic fire or explosion at the HTEF to initiate an accident previously evaluated in the UFSAR is addressed in the response to Question 5.

Question 2: Does the proposed activity result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR?

The new steam, condensate, demineralized water, and high-voltage electrical equipment are not themselves SSCs that perform a function that is important to safety. The performance of the new equipment under normal operation or under anticipated operational occurrences has minor effects on plant systems (as discussed in the response to Question 1) and does not result in the malfunction of SSCs important to safety. (Note that significant transient events or conditions are tracked under an existing station program to ensure that fatigue limits on SSCs are maintained.)

The installation of new steam, condensate, and demineralized water piping and components and the installation of new high-voltage electrical equipment introduces the potential for a catastrophic failure in any of these to affect SSCs important to safety. As discussed in the response to Question 1, the codes, standards, and practices used in the design, construction, and operation of such items will provide sufficient assurance that the likelihood of a catastrophic failure is very low. In addition, the routing of the piping and location of new components minimizes or eliminates the potential for a catastrophic failure to cause a malfunction of an SSC important to safety. Therefore, there is no more than a minimal increase in the likelihood of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The proposed activity includes the installation of several new instrumentation and control devices at the main control board. Operation or mis-operation of these instruments and controls cannot cause a malfunction of an SSC important to safety, and the new devices will not interfere with the operation of existing equipment in the main control room. This is assured because the new simplified steam HTEF controls are not integrated into the nuclear power plant control systems and thus any failures of new controls are not capable of impacting existing systems with ties to SSC functional controls. Select information from the new equipment and controls is available in the main control room; the routing of this information will not adversely affect existing information systems in the main control room.

The potential for a catastrophic fire or explosion at the HTEF to cause malfunctions of SSCs important to safety is addressed in the response to Question 5.

Question 3: Does the proposed activity result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR?

As discussed in the response to Question 1, the performance of the new equipment under normal operation or under anticipated operational occurrences has minor effects on plant systems and does not result in any accident that has radiological consequences.

The installation of new steam piping and components introduces the potential for a failure in any of these that could result in a high-energy line break in the plant. The existing plant systems include steam piping and associated components whose failure could have radiological consequences. The bounding analysis for such events is the analysis of the radiological consequences of the rupture of a main steam line. That analysis assumes that primary to secondary leakage is released directly from the faulted steam generator to the environment and also through the power-operated relief valves of the unfaulted steam generators. The analysis remains bounding for the case of a rupture in the much smaller steam lines to be installed under the proposed activity.

The installation of the new equipment does not affect the response of SSCs credited for accident mitigation. The installation of the new equipment, including new high-voltage electrical equipment and associated protective relaying to provide power to the HTEF, does not affect the post-accident response of the main generator or plant electrical systems.

Control failures and the potential for new equipment response impacts are addressed in the response to Questions 1 and 2.

Therefore, the proposed activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

Question 4: Does the proposed activity result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR?

The installation of new steam, condensate, and demineralized water piping and components and the installation of new high-voltage electrical equipment does not introduce the possibility of a change in the consequences of a malfunction because the new equipment is not an initiator of any new malfunctions of SSCs that could lead to or mitigate radiological consequences and no new failure modes of such SSCs are introduced.

Control failures and the potential for new equipment response impacts are addressed in the response to Questions 1 and 2.

Therefore, the proposed activity does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Question 5: Does the proposed activity create a possibility for an accident of a different type than any previously evaluated in the UFSAR?

General Design Criterion 4 in Appendix A to 10 CFR Part 50 includes a requirement that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions and dynamic effects associated with postulated accidents that may result from events and conditions outside the nuclear power unit. The HTEF, which will be producing hydrogen nearby, introduces the potential for an explosion or fire at a nearby facility to affect SSCs onsite.

Evaluations performed for the siting of a generic HTEF assumed that the HTEF electrolyzers would be located at least 500 m from SSCs important to safety [Reference 1]. This minimum distance represents the point at which the overpressure effect from the maximum credible accident at the HTEF would fall below 1 psi. At this distance, a deflagration event is expected to have no consequence to the nuclear plant.

The UFSAR for the reference plant used in developing this report discusses the potential for an explosion at a nearby facility or on a nearby transportation route. An evaluation compared the probability of an explosion to the acceptance criteria of Regulatory Guide 1.91 ("Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants"), Rev. 1 [Reference 2] and concluded that an explosion on a nearby route was not a credible event.

Revisions 1 (February 1978), 2 (April 2013), and 3 (November 2021) of the current version of Regulatory Guide 1.91 each provide: (1) a method for determining the distance from critical plant structures to the location of the explosion, a distance beyond which any explosion is not likely to have an adverse effect on plant operation or prevent a safe shutdown, and (2) methods for determining whether the frequency of occurrence of an explosion is sufficiently low. Although there are differences between these various versions of the Regulatory Guide, the basic methodology for determining the distance beyond which any explosion is not likely to have an adverse effect on plant operation or prevent a safe shutdown has not changed.

An evaluation of the explosion hazard from the proposed HTEF was not performed. However, based on the amount of hydrogen that would be released from a failure at the HTEF, it is expected that such an evaluation would show that the relevant Regulatory Guide 1.91 criteria for the distance from the HTEF to SSCs important to safety is met. Therefore, the presence of the HTEF would not create a possibility for an accident of a different type than any previously evaluated in the UFSAR. This is supported by the conclusions of the explosion hazard analysis performed in Reference 1. Reference 1 analyzed an HTEF with hydrogen volumes bounding the proposed pre-conceptual design; this analysis determined that a 500 m standoff distance to the most fragile nuclear SSCs, the switchyard transmission towers, is viable.

Operator errors involving the new operator controls for steam and electric power to the HTEF could initiate, at most, small transients in balance-of-plant systems. As discussed in the response to Question 1, changes that might result in these small transients are within the normal operational capabilities of the turbine control system and the other plant control systems. Such transients do not reach the threshold of an "accident previously evaluated in the FSAR." In addition, such minor transients do not constitute an accident of a different type.

Control failures and the potential for new equipment response impacts are addressed in the response to Question 1 and Question 2.

Therefore, the proposed activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR.

Question 6: Does the proposed activity create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR?

As discussed in the response to Question 2, the installation of new steam, condensate, and demineralized water piping and components and the installation of new high-voltage electrical equipment introduce the potential for a catastrophic failure in any of these to affect SSCs important to safety. The new piping and components will be located such that there is no potential for a catastrophic failure to cause a malfunction of an SSC important to safety with a different result (e.g., failure of redundant SSCs important to safety) than any previously evaluated in the UFSAR.

As discussed in the response to Question 5, it is expected that an evaluation of the explosion hazard from the proposed HTEF would show that the relevant Regulatory Guide 1.91 criteria for the distance from the HTEF to SSCs important to safety is met. Therefore, the presence of the HTEF would not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

Question 7: Does the proposed activity result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered?

The installation of new steam, condensate, and demineralized water piping and components and the installation of new high-voltage electrical equipment introduces the potential for failures of these new SSCs to produce small changes in the flow of steam, condensate, and demineralized water, and also changes to the electrical power system. Such changes are not associated with any fission product barrier. Therefore, the proposed activity does not result in a design basis for a fission product barrier as described in the UFSAR being exceeded or altered.

Question 8: Does the proposed activity result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses?

The methods of evaluation used in determining the effects of a transient in the cold reheat or other extraction steam systems are not described in the UFSAR. The methods of evaluation used in determining the effects of a transient in the plant electrical systems are not described in the UFSAR for the reference plant.

Relatively small quantities of hydrogen are currently stored at the site for use in the main generator and in the chemical and volume control system. No methods of evaluation for determining the impact of a hydrogen fire or explosion on site are described in the UFSAR.

The proposed activity involves the production and transportation of larger quantities of hydrogen near the site. As discussed in the response to Question 5, it is expected that an evaluation of the explosion hazard from the proposed HTEF would show that the relevant Regulatory Guide 1.91 criteria for the distance from the HTEF to SSCs important to safety is met. Because such an evaluation would use the same method of evaluation used in Regulatory Guide 1.91, the method of evaluation would be one approved by the NRC for the intended application.

Therefore, the proposed activity does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

4. CONCLUSION

Generic 10 CFR 50.59 evaluation of the pre-conceptual design described in Reference 3 concludes that a 500 MW_{nom} high-temperature electrolysis facility addition to a nuclear power plant can be performed under the 10 CFR 50.59 process for many plants within the existing U.S. nuclear fleet. This was determined based on the limited impact of the hydrogen production facility on the reference plant mechanical, electrical, and controls systems.

This generic evaluation assumed the presence of an existing analysis for explosive hazards in the vicinity (i.e., explosions at nearby facilities or on nearby transportation routes) of the nuclear plant. Hazard analysis would need to be performed to assess whether there is (1) the possibility for an accident of a different type than any previously evaluated in the UFSAR and (2) the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR. Nuclear plants that do not have an existing hazard analysis meeting these requirements may need to pursue a license amendment request.

This evaluation is specific to the facility design and reference plant documented in Reference 3, and cannot be used as a template for other nuclear power plants. Consultation with the nuclear steam supply system and turbine generator vendors should be performed as needed to validate plant responses for the various failure scenarios considered in this evaluation. Other plants may have more detailed discussion of methods of evaluation, additional accidents or transients considered in their UFSAR, and unique design features which would need to be addressed in the project-specific evaluation responses.

5. REFERENCES

1. INL/EXT-20-60104, Rev. 1, *"Probabilistic Risk Assessment of a Light Water Reactor Coupled with a High-Temperature Electrolysis Hydrogen Production Plant,"* Vedros/Christian/Rabiti, November 2022.

2. RG 1.91, Rev. 1, "*Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes Near Nuclear Power Plants,"* U.S. Nuclear Regulatory Commission, February 1978.

3. SL-016181, Rev. 1, "Nuclear Power Plant Pre-Conceptual Design Support for Large-Scale Hydrogen Production Facility," Sargent & Lundy, November 2022.

Appendix B

Design Case Assumptions

(Figures and numbering are from S&L Reports)

B-1. Reference NPP Assumption

- Westinghouse 4-loop PWR rated at 1200 MWe nominal.

B-2. Electrical Requirements

- 105 MWe AC power
- Assumed 10% Auxiliary Power (approximately 10 MWe) for Hydrogen Generation Plant
- The electrical system demarcation between the NPP and H_2 Generation Plant Design will be at the low (medium voltage) side of the step-down transformer to the H_2 Generation Facility
- Electrical distribution design required for the H_2 Generation Plant beyond the demarcation is assumed inclusive of the H_2 Generation Plant Design
- Modeling tools will confirm acceptable NPP electrical integration basis and voltage drop acceptability of power take-off to the demarcation.

B-3. Thermohydraulic

- Steam Input Requirement to the Hydrogen Island
  - 300°F steam at 50 psig at the H_2 Plant Demarcation Border
  - The value chosen is assumed to be bounding for the different H_2 system technologies.
- 25 MW thermal power extraction from the steam cycle
  - This value accounts for the projected need by the HTE equipment as well as the system losses due to the delivery from the NPP.
- All cooling water needs would be inclusive of the H_2 Generation Plant Design
  - No cooling water will be supplied from the NPP Generation Facility.
- Heat balance modeling will preliminarily confirm secondary plant impacts associated with steam extraction.

B-4. Hydrogen Generation Plant

- Plant Rating
  - Nominal 100MW (105 MWe and 25 MW_{th})
  - Hydrogen production approaching 60 metric tons per day.
- Location
  - 0.5 to 1 km from the NPP.

B-5. Electrical Design

- High-voltage electrical source targeted at the high side of the existing main power transformer
- Isolation device located at the NPP
- High-voltage transmission line to Hydrogen Facility
- Isolation device located at the Hydrogen Plant
- Step down transformer located at the Hydrogen Plant
- 10% margin will be included with the electrical feed to account for hydrogen plant auxiliary power needs.

Figure 11 illustrates the pre-conceptual design for the H_2 plant electrical feed scope.

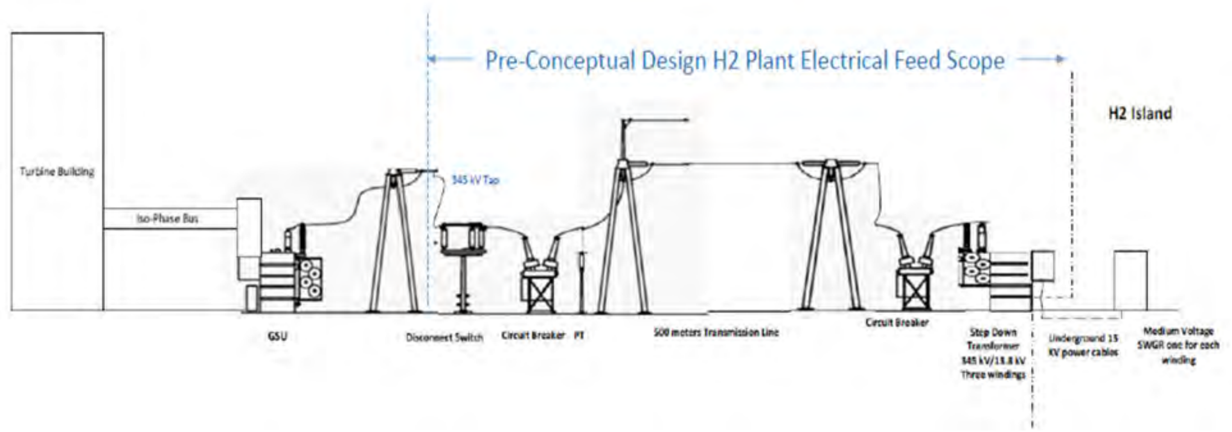


Figure 11. Pre-conceptual design for the H2 plant electrical feed scope

B-6. Thermohydraulic Design

- Extraction point is targeted at crossover piping between the high-pressure turbine and the MSR.
- Piping connections from the crossover piping to new reboiler.
- Addition of a flow-limiting device (if determined to be necessary) to limit flow assumed in safety analysis for inadvertent opening of a main steam relief valve or single steam dump/steam bypass valve.
- Reboiler condensate return to the main condenser.
  - Potential usage of a demineralized water pre-heater on the reboiler condensate discharge
- Demineralized water supply to reboiler
  - Potential usage of reboiler feed pump.

Figure 12 depicts the thermohydraulic design for the turbine building.

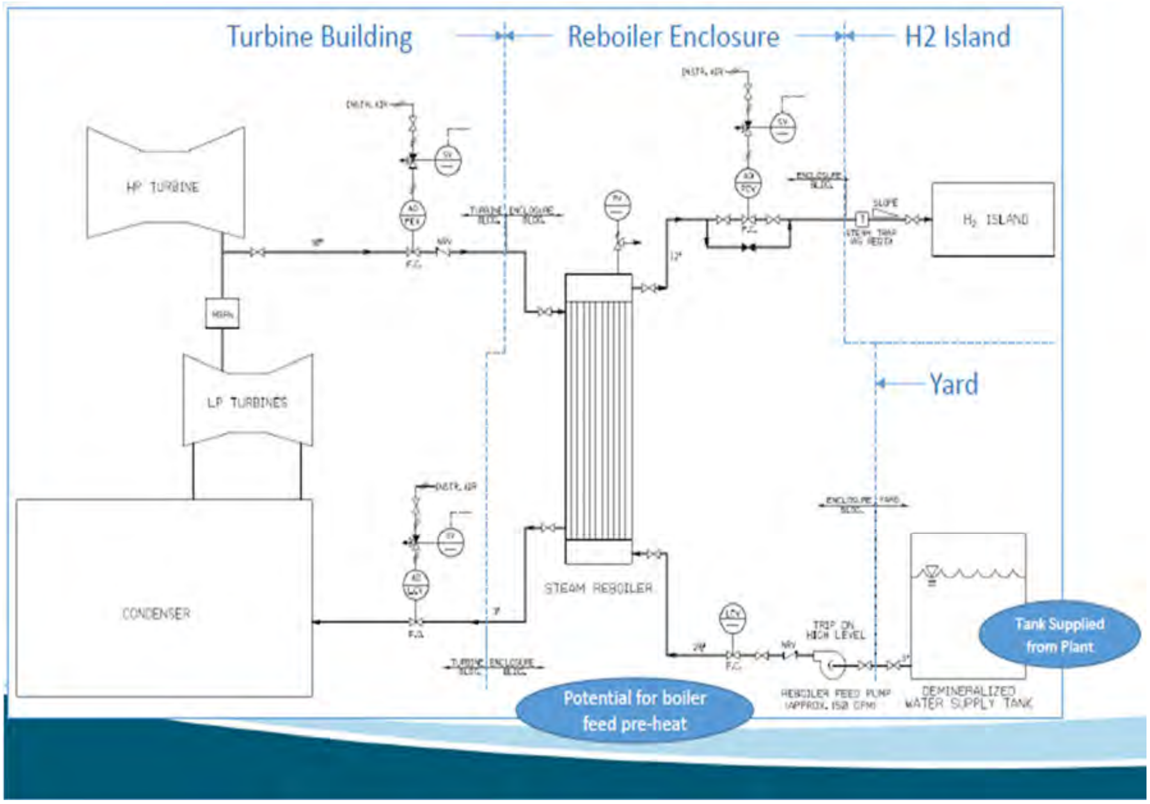


Figure 12. Thermohydraulic design for H_2 system.

B-7. Control Systems Design

The conceptual design includes three control areas to be considered, and they are:

- Area 1 is applicable to both local and Main Control Room (MCR) control interfaces for the extraction of steam and the supply of electricity to the HTE hydrogen plant.
- Area 2 is applicable to the control interfaces for new plant reboiler added to support the steam extraction and energy conversion prior to delivery to the HTE hydrogen plant.
- Area 3 is applicable to the interfaces and controls for HTE hydrogen plant.

Figure 13 shows the three areas that are being considered.

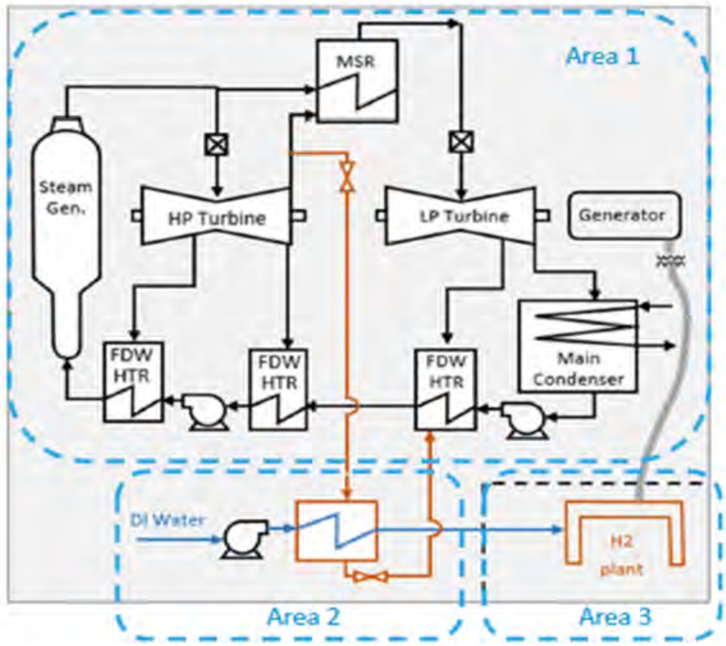


Figure 13. High level HTE plant layout.

(No material changes made in Revision 2)

Appendix C

Advisory Subcommittees Excerpts

C-1. Internal/External Events and PRA Subcommittee

PRA Subcommittee laboratory and operational expert review involved siting considerations to prevent adverse plant operational effects on safety related systems, structures, and components (SSCs) due to the addition of a heat extraction system and the potential for hydrogen detonation failure scenarios that are postulated to occur based on breach of HTE equipment pressure boundaries.

The changes to, and the possibility of additions to, internal and external events were considered in two reports in 2020, "Probabilistic Risk Assessment of a Light Water Reactor Coupled with a High-Temperature Electrolysis Hydrogen Production Plant" and "Final Report on Hydrogen Plant Hazards and Risk Analysis Supporting Hydrogen Plant Siting near Nuclear Power Plants." A high-level summary of the PRA methodology follows, followed by a medium-level overview of the PRA methodology and results.

The generic PRAs consisted of the following methodology:

- Safety logic modeling of two designs of heat-extraction systems (HES), leak detection and mitigation at the high-temperature electrolysis facility (HTEF), detonation at the HTEF, direct coupling of electricity between the NPP and HTEF, and additional effects impacted to existing fault and event trees.
- Hazards analysis, including an FMEA for all designs modeled.
    - The top two were an unisolable main steam line break in the HES and increased LOOP frequency from a detonation event at the HTEF.
- Identification and impacts of jurisdictional boundaries.
- Preliminary example plant site considerations.
- Identification of NPP critical structures and their fragilities to overpressure events.
- Some key assumptions made for the state of the generic model:
    - HES isolation is accomplished through the same MSIV configuration as the NPP uses.
    - Distance from the detonation to the NPP critical structures is 1 km.
        - Sensitivity study is performed for minimum safe distance.
    - The HTEF will not store product hydrogen on site.
    - The storage facility is 5 km distant from the NPP critical structures.
    - Detonation frequencies are determined by one proposed HTEF module's piping and instrumentation diagrams.
    - Overpressure experienced at distance determined by available hydrogen in two scenarios:
        - High pressure jet leak detonation
        - Cloud detonation.
- Determination of increased initiating event frequencies for existing design basis events.
- Determination of any new initiating events (none).

- Determination of increased CDF and LERF.
- Comparison of results to criteria in 10 CFR 50.59.

The following is a medium level overview of the generic PRAs [2].

The generic PRAs start with the safety logic modeling of the HES designs. The HES will be integrated with the NPP main steam at an outlet downstream from the NPP's main steam isolation valves (MSIVs). At the time of the PRA report, there were three designs considered for the HES. The first design was a two-phase-to-two-phase transfer design where the heat-transfer steam is tapped before entry into the turbine and the thermal power delivery (TPD) loop enters a vapor phase when heated to operating temperatures. The second design was a two-phase-to-one-phase transfer where the heat-transfer medium stays in the liquid phase. The third design was a two-phase-to-two-phase transfer design where the heat-transfer steam is tapped after the first turbine, then sent to a reboiler where the TPD loop enters a vapor phase when heated to operating temperatures. Steam-to-steam heat transfer will always use the two-phase-to-two-phase design. Heat-transfer fluids (HTFs), many times incorrectly referred to as "heating oil," were characterized but not evaluated for probabilistic effects. HTFs can be used in two-phase or single-phase operating states depending on their physical characteristics and the desired operating temperature. Note that there was no actual HES system at the time of this research, and therefore, these were conceptual designs. A two-phase-to-two-phase design is the more likely of the two systems, given the advantages and familiarity of using steam; therefore, it was conservatively assumed for the probabilistic analyses. The analysis resulted in an increase in the existing initiating event frequency for an un-isolable main steam line break, among other considerations.

Jurisdictional boundaries were considered for licensing pathways. The NRC was found to have jurisdiction up to and including the site boundary. Most events that can interfere with the operation and safety of the NPP affected by the location of the HTEF outside of the regulatory jurisdiction are treated as external events. The exception is the potential of reactivity feedback that would occur if there were a sudden large leak in the TPD loop that services the HTEF.

The HES design options and assumptions considered for the representative NPP, HES, and HTEF are listed in this report. Some key assumptions are as follows:

1. The HES isolation valves are in the same configuration as the NPP's MSIVs.
2. Steam is the heating medium.
3. Production hydrogen will be piped to a storage facility 5 km distant.
4. Electrical power linkage between the NPP and HTEF will be through the grid to buffer direct upsets.
5. The HTEF is 1 km from the nearest NPP critical structure.

The reactor building and other critical structures external to the reactor building (e.g., coolant storage tanks) were evaluated for fragility to an overpressure event. The PRA report lists in detail the assessments drawn upon for the blast fragility analysis. Most fragility analyses relied on the conversion from blast overpressure to published structural wind fragilities. Missile fragilities were also adapted from published wind missile studies. By far, the most susceptible components that would affect an existing initiating event at a NPP were the switchyard components. Loss of switchyard components means a loss-of-offsite-power (LOOP) event, which challenges the NPP to shut down safely. Switchyard components are fragile to wind pressure, and therefore, are also fragile to an overpressure event.

The hazards associated with the addition of the HES designs to the existing NPP were considered through interviews with subject matter experts (SMEs), available design drawings, and options of the proposed HES. A FMEA was performed, and a comprehensive list of hazards were identified and considered for inclusion in the PRA. A sampling of key hazards initiating events either affected or added to the PRA by the addition of the HES and the HTEF are as follows:

- Overpressure event effects on LOOP, loss of service water, and critical structure damage
- HES steam pipe rupture outside of NPP MSIVs causing a main steam line break
- Prompt steam diversion loss feedback from TPD loop rupture.

Detonation frequencies were determined by a bottom-up analysis of leak frequencies associated with the proposed HTEFs plumbing and instrumentation diagrams. The Sandia National Laboratories (SNL) report details the leak frequency analysis. The leak frequencies per year were converted into a detonation frequency per year of operation. A potential detonation-causing leak was determined to occur at a frequency of 5.2E-02/y for an 1150 MW_{nom} HTEF or at a frequency of 4.6E-03/y for a 100 MW_{nom} HTEF.

Two types of potential detonations were identified: a high-pressure jet of hydrogen or an accumulated cloud of hydrogen. The bounding case of overpressure for both types of detonations at 1 km is shown in Table 2. The most susceptible component is the transmission tower in the switchyard with a 0.8 probability of failure at 0.2 psi.

Table 2. Consequence results from risk analysis.

Detonation Type	Bounding Overpressure at 1 km (psi)
Jet Ignition	0.06
Cloud Ignition	0.4

Hazard evaluation was performed by INL for the PRA. No credit was given for attenuation of the shock wave made by buildings, wooded areas, or other topography. The bounding case used the largest leak size, denoted 1.0, and therefore, this frequency (5.2E-02 /y) was used in the PRA IE development. Calculations were made for the next largest leak size, denoted 0.1, and the most fragile component of the NPP (transmission tower) was not affected by the overpressures created from either the high-pressure jet or hydrogen cloud detonation.

High-pressure jet detonation occurred at a frequency of 1.82E-02 /y for the large HTEF and 1.60E-03 /y for the small HTEF. A cloud detonation occurred at a frequency of 4.2E-9/y for the large HTEF. The overpressure consequence at 1 km for the jet ignition was not enough to damage the transmission tower. The frequency of the cloud detonation for the bounding large HTEF was 7 orders of magnitude under the current LOOP frequency and 5 orders of magnitude under the current loss of service water frequency. Therefore, hydrogen detonations are effectively screened out as hazards at 1 km. A sensitivity study was performed on the jet ignition distance to the transmission tower and a safe distance was determined at 500 m.

A large steam line break is the most common hazard introduced by adding the HES to the NPP. There are two isolation valves for the HES, set in a series configuration. The success of these valves is the first line of defense of a steam line rupture within the HES after the NPP's main steam MSIVs. Rupture of the isolation valves were also modeled. After the isolation valves, all of the other active components in the Piping and Instrument Diagram (P&ID) were evaluated in the fault tree of the HES, which was added to the MSLB event tree and associated affected trees of the NPP PRA model.

Two generic NPP PRAs were prepared—one was a PWR and the other was a BWR. To remain generic, external events other than those created by the addition of a HTEF near the NPP were not included in the model. A Mark I containment BWR and a four-loop PWR were modeled. From the generic PRA starting point, modifications were made to the internal event logic models for cases where systems might be affected by the addition of the HES and the HTEF. External events were considered to be a result of an HTEF hydrogen detonation and was represented by an increase in the switchyard centered LOOP frequency. The external hydrogen detonation event was also analyzed for inclusion in the PRA on its own as potentially damaging to critical structures.

For the bounding large HTEF, PWR PRA results showed the initial initiating event frequency for main steam line break (MSLB) went up by 5.6% from the initial value. The switchyard related LOOP initiating event frequency increased by 1%. The bounding large HTEF with PWR CDF increased a minimal 6.56% from 8.33E-06 to 8.88E-06 /y.

BWR PRA results were even better. The addition of steam line break IE frequency to the existing general transient initiator is trivial (added 0.002%). The IEs related to a switchyard-induced LOOP are the same as the PWR model because such events are indifferent to the reactor types. BWR CDF did not change to two significant digits, before and after at 2.84E-05 /y.

A reference study commissioned by INL noted that nearly all criteria are readily met for a modification, such as the HES through 10 CFR 50.59, except there were not enough data to determine if the minimal increase in design basis accident (DBA) frequency is met. The study noted that this minimal increase is traditionally understood to be less than 15%. The study recommended further PRA evaluation to determine the DBA frequency impact. This subsequent PRA study found the largest increase in a DBA yearly IE frequency to be 6% (large steam line break for the PWR), thus meeting the criteria for 10 CFR 50.59.

If the Regulatory Guide 1.174 approach is used to bolster the consideration of the plant modification, one of the decision metrics is the risk associated with proposed changes in plant design and operation. Specifically, thresholds and guidelines are provided for comparison with Level 1 PRA results for CDF and LERF.

As described in Regulatory Guide 1.174 and shown in extracted Figure 14, CDF should be below 1E-5 overall and the change in overall CDF should be below a magnitude of 1E-5. Any plant, which starts at a 1E-4 or more CDF, requires less than 1E-6 increase in CDF to be considered. If these metrics are met, NRC most likely considers this a small change, which is consistent with the intent of the Commission’s Safety Goal Policy Statement and a detailed quantitative assessment of the base values of CDF is not necessary for the license review.

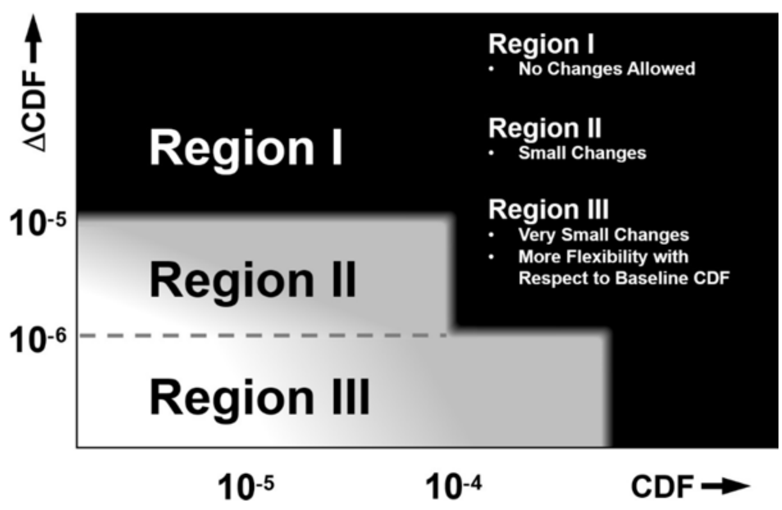


Figure 14. Acceptance guidelines for CDF.

The generic PWR and bounding large HTEF considered for this study has a nominal CDF of 8.33E-06 /y, and the increase after the addition of the HES and HTEF is to 8.88E-06 /y for ΔCDF of 5.50E-07 /y, which is well within Region III of the acceptance guidelines shown.

The generic BWR and bounding large HTEF considered for this study has a nominal CDF of 2.84E-05 /y, and the increase after addition of the HES and HTEF is still 2.84E-05 /y for ΔCDF of 1.00E-07 /y, which is well within Region III of the acceptance guidelines.

Interaction between the PRA subcommittee and the INL PRA team pointed out the following benefits and potential improvements of the generic PRA for use as a tool for site-specific licensing support:

- The existing SAPHIRE PRA model can be shared as is. However, it would be beneficial to translate the existing SAPHIRE PRA model to a prominent industry Computer Aided Fault Tree Analysis system (CAFTA) model.
- It is important to modify the model to match the current design of the HES and the HTEF as details become available.
- Consider hydrogen detonation seismic induced failures.
- Model the effects of site layout and the natural and engineered barriers on hydrogen detonation effects and use this information to site an HTEF.
- Model a complete HTEF with interconnecting piping and storage for use in the next PRA.

C-2. Integrated Operations and Reactor Impacts Subcommittee

The Integrated Operations and Reactor Impacts subcommittee informed AE design activities related to integration of hydrogen electrolysis within an existing nuclear facility to prevent unexpected or adverse plant operational effects.

Pre-eminent among design principles is that electrolysis equipment integration does not translate adverse control effects back to the nuclear reactor. It follows that this is also a fundamental premises to be demonstrated in regulatory evaluation performed under the 10 CFR 50.59 or license amendment review processes for any modification to an operating nuclear facility. The Institute of Nuclear Power Operations coined the concept of maintaining a *line-of-sight to the reactor core* with respect to managing reactivity through operational crew performance, teamwork, and fundamental operator behaviors and knowledge. This principle is also aligned with licensing approval requirements that apply to modifications to the nuclear plant. Modifications in support of nuclear-integrated the must be designed from first principles to not result in normal, upset, or transient conditions, which could challenge the operating crew’s ability to control the reactor as required by 10 CFR 50.59.

Operations subcommittee critical review areas based on the potential to influence the first phase of this 100 MWe/25 MW_{th} nuclear-integrated HTE hydrogen demonstration project design and 10 CFR 50.59 research deliverable:

1. Subcommittee Review Approach:

- Perspective of operations impacts that include reasonable operator action with zero impact to line-of-sight to the reactor core and scenarios that could affect existing accident scenarios or could conceivably introduce new scenarios
- Operational modes with varying degrees of operator involvement and the corresponding impacts on the Licensing Bases (Operating License and UFSAR)
- Communications protocol for integrated energy systems dispatch with the local balancing authority, the utility, nuclear facility, and among the units for a multiple unit site are also presented.

2. Potentially operations-sensitive design assumptions associated with the 100 MWe/25 MW_{th} integrated nuclear hydrogen demonstration project, including critical input area considerations:

- Transparent operation (focus remains on the core)

- System indication of control system operation to inform operators, but a separate and independent system operator is controlling hydrogen production
- Turbine control system to maintain secondary heat load
- Effect of concurrent power changes with hydrogen system operation
- Requirements to buck/boost
- Communications with Operations, Plant Staff, and ECC
- Water chemistry impacts
- Secondary chemistry impacts
- Punch out function/automatic drop-off, including effect during an independent transient
- Thermo-hydraulic, electrical power, and operational control design elements
- The addition of an indirect in-plant steam reboiler (steam generator) and condensate return system:
  - Steam off-take thermo-hydraulic design and control logic
- Extraction of 25 MW_{th}, 300°F + saturated steam:
- MSR crossover piping fed (Design Option 1)
  - Integrated energy conversion heat exchange reboiler with plant steam-side heating of demineralized water-to-steam for feed to the hydrogen island electrolyzer skid
  - Level controlled reboiler condensate return to the secondary plant drain system
  - The possible effects of normal, upset, and transient stream extraction effects on reactor temperature and reactivity changes, including secondary plant and extraction drain system stability.
- The electrical powering methodology for the integrated electrolysis plant equipment:
  - Behind-the-meter high voltage AC electrical off-take and connection design
  - Stepdown AC transformers and protective relaying
  - DC rectified power feeds and control
- New equipment control scheme approaches compared current licensed plant designs.

3. Specific operational control assumptions:

- Third-party operator-controlled hydrogen island concept that is dispatched through the plant's MCR with hydrogen island equipment operating locally outside the plant protected area
- The individual and integrated licensed, field, and hydrogen island operator command and control roles, responsibilities, and teamwork impacts
- Startup, maneuvering, and shutdown operational control assumptions with third party dispatch interface procedures with the control room for thermal and electrical power to the hydrogen island
- Likely dispatch limitations and transitions between electrical and hydrogen production
- Cursory review of practical hydrogen safety aspects of the operational design
- This controls scheme needs to employ permissive startup features and emergency isolation capabilities

- Procedural and timeliness considerations of establishing plant readiness for hydrogen dispatch, including steam, condensate return, power, and demineralized water to support hydrogen island operations
- The MCR can shut down the hydrogen island equipment for any reason by isolating the extraction steam control valve from the MCR
- Feedback related to new areas that will be included in operator training programs.

4. Operations Environment:

- Operation of the Integrated Energy Systems (IES) should be transparent to the Operations Control Room staff allowing the focus to remain on the core and attendant power production functions.
- Procedure guidance should be implemented to provide for and limit impacts on operator action to a pre-planned diversion of steam from the turbine to the IES heat exchanger (lower turbine load by approximately 1.5%).
- Operator actions should be assumed to be limited to the turbine control adjustment and should be based on plant design and temperature control to maintain operation within reference bounds.
- The current demonstration scale (100MW) requires approximately 25MW of thermal power to assist in steam generation for the HTE process flow, which should have little-to-no observable impact on reactor temperature with constant steam flow.
- The alignment will proceed with concurrence by station personnel and with approval from the shift manager as described below.
- IES operation and performance indication should be available to the Operations Control crew.
- A separate and independent hydrogen system operator should remotely control the electrolyzer and balance of plant operation and production.
- The control room operators retain the ability to reject the steam flow to the reboiler (redundant isolation) and restore normal plant alignment.
- The operators will respond to consequential demands for reactive power given the (small) addition of inductive load (or loss).

5. Control Systems – Operational Design Considerations:

- System indication should be provided on the MCR board with auxiliary steam header pressure.
  - A human-machine interface (main and back-up) with a screen for monitoring and capability to terminate function is assumed.
- Transient during IES operation should be addressed.
- The steam supply should have the option to be operated from the MCR.
- Conscious consideration of whether automated non-class control systems, which cause a turbine runback, setback, etc., cause an automated isolation of the steam supply to HTE?
    - Might not be necessary since turbine will runback/setback based on primary and secondary power mismatch.
- The system should be started up/shut down via automation, if possible.
- The system should have an automatic isolation feature based on steam break downstream of supply valve.

6. Steam – Operational Design Considerations:

- Steam supply from the main steam header, if used, should be downstream of the main steam isolation valves.
- Steam should isolate automatically (or quickly/easily manually) upon a plant transient or reactor trip to minimize effects of plant cooldown.
- Depending on load, ideally, steam demand will be a constant source/load when set to minimize thermal power or secondary plant swings.
- Preconceptual design shows the H_2 plant is nominal 100 MW (105MWe and 20MWth). 20 MWth is ~.6 to.8% in most PWRs.
- Return to main condenser—steam line leaks and Radiation Protection issues.
- SG health—design related issues but needs to be considered because of the addition of the piping (iron transport).

7. Electrical

- Electrical modeling would need to adequately assess a full load reject of 105 MWe and impact on turbine/generator speed/frequency, protective and controls for generator/excitation main transformer, grid breakers, etc.
- Given the periodic examples of remote grid loss of load impacts on nuclear unit trips [12], a loss of 100 MWe directly at the site needs to be evaluated.
- North American Electric Reliability Corporation (NERC) requirements will play a role in how the electrical tie-in is designed and how the generator output is protected.

8. Communication Protocols MCR/NPP Load Dispatcher

- Each nuclear utility and plant will establish communication protocol to declare times of availability to support HTSE operation.
- Economic vs. baseload operation of the HTSE systems will dictate the operation mode and the communications protocol.
- For baseload operations, utility decisions to engage the HTE will be based on demand and market conditions and will be initiated by grid price triggers.
- Notice to the regional balancing authority should be made in advance of a request to the nuclear facility/unit that has previously declared that it is available. Units performing required surveillance testing, power operations in response to grid related support, or are critical in high-load conditions will not opt in. Availability may also be based on time in core life, although the potential impact on core reactivity is significantly smaller than power derates associated with economic load dispatch.
- The actual process for communications must also be established and is subject to data system security and should be considered with respect to cost impacts to automate the function.
  - As an example, the Byron Station and the fleet of Babcock & Wilcox (B&W) plants have dedicated monitors in the control room that (can) identify requests for economic dispatch.
  - Since operation of the HTE is not expected be in a load-follow mode (either on or off for extended periods of time), manual notification from the balancing authority to the utility and plant seems more appropriate and sufficient. Details will need to be evaluated as to how incorporate HTE operation into the work control structure.

C-3. Electrical and Switchyard Subcommittee

The purpose of the Electrical and Switchyard Subcommittee work was to ensure any new electrical power aspects that is needed to support HTE equipment within an existing nuclear facility will not create unexpected or adverse plant operational effects.

Early considerations identified to inform design/10 CFR 50.59:

- The HTE equipment's requirement for sufficient electrical supply from power plants will require modifications to the NPP's electrical infrastructure.
    - Those modifications require modification type 10 CFR 50.59 screening to ensure proper implementation reviews and documents are completed.
    - Many installation aspects of the electrical portions of the proposed equipment will be categorized as "Commercial or Equivalent" type installations as defined by the industry generic procedure IP-ENG-001.
    - Within this commercial modification process, the requirements for implementors to be cognizant of certain aspects of the infrastructure expansion, which require review under 10 CFR 50.59 prior to implementation, will need to be met.
- Generically, station UFSARs describe in detail many of the critical parameter of the Main Power Generator System that the HTE equipment is proposed to interconnect with.
- Parameters, including generator capacities, short circuit, voltages, grid interconnection descriptions, iso-phase bus duct ratings, generator step up transformer parameters, transmission line parameters, and grid fault contributions, are expected to change and could be subject to 10 CFR 50.59 review.
- Many nuclear plant facilities take advantage of main generator power during the initial moments of a Loss-of-Coolant Accident (LOCA) prior the main generator tripping offline.
- Safety related systems are often started from the main generators source then fast transferred to other offsite sources upon the main generator trip.
    - As the HTE project interconnects electrical connections and components in these areas, the licensee shall review adverse potential aspects of adding these connections and components under the 10 CFR 50.59 review (even possibly NRC approval prior to implementation) dependent on each station's licensing commitments related to the main generator's description in the UFSAR.
- Each station shall expect to require UFSAR revisions related to the proposed electrical infrastructure expansion. Topics recommended to be added into the UFSAR include, but is not limited to, description of new components and connections, description of new connections and component's automatic electrical protection schemes, revisions to critical parameters post design analysis results (short circuit, voltage, capacities), operations interface, and procedural description describing the HTE equipment operation.
- The electrical infrastructure operation support is also a topic likely requiring 10 CFR 50.59 review if the Nuclear Operations team will be required to operate and monitor the new components and connections. Under 10 CFR 50.59, the licensee will likely be required to review the added activities to the operators including these:
    - All interfaces between new or modified equipment and the station electrical systems should be identified.
    - The effects of possible electrical transients (e.g., switching or faults) on the station systems that could occur as a result of new or modified electrical equipment should be identified.
    - The expected operator interfaces and interactions with new or modified electrical equipment should be identified on the operator interface.

- Design considerations to include auto-operation elevating operations loading.
- Operations watch loading considerations.
- Modeling of system in station simulator.
- The design must ensure that for new or modified electrical equipment and devices all applicable design and functional requirements (including applicable codes, standards, etc.) for the affected station electrical systems continue to be met.
- Fire impact, security plan, and lighting.
- Fire protection review.
- Security review impact and contingencies.

  Lighting plan within security requirements.

C-4. Control Systems Subcommittee

The purpose of the Control Systems Subcommittee is to ensure that all new control aspects for plant operational control of HTE equipment within an existing nuclear facility will not create unexpected or adverse plant operational effects.

The control systems conceptual design was evaluated from both the mechanical and electrical aspects.

- Key elements of the mechanical design included these:
  - One of more air operated valve(s) (AOV) for the steam inlet with position indication
  - Steam line pressure indication
  - Manual valves for the condensate return back to the NPP
  - Local air system instrumentation.
- The assessment of the electrical aspect primarily focused on breaker/disconnect position indication.

Controls review input focused on the importance of simplicity within the control scheme and conformation that the valve closure times are evaluated for any pressure transients that might occur on the associated steam system.

The overall control scheme for the system is assumed to be comprised of only controls within the MCR to initiate the flow of electricity and steam to the hydrogen plant.

- With a larger scale (100 MW) hydrogen plant, it is assumed that the plant is operated by personnel that are different than those operating the nuclear plant.
- Controls for the hydrogen plant itself are not part of this control scheme.
- It is assumed that an intermediate heat exchanger (reboiler) is used to separate the nuclear steam from the hydrogen plant.
  - This would eliminate the need for any type of radiation monitoring.
  - Condensate from this heat exchanger is returned to the plant.
- The hydrogen plant only requires electricity, steam, and demineralized water to operate.
- To keep controls as simple as possible, only a few items are needed for the MCR operator.
- For the electrical inputs to the plant, controls that are typical for operating switches/disconnects/breakers to initiate the flow of electricity are included such as control switches and position indicating lights. Typical alarm indications would be included.
- As for the steam inputs, controls that are typical for the valve operation are assumed.
- Control switches and position indication for an air operated valve are included.

- Typical alarm indications would be included.
- Based on conversations with electrolyzer vendors, pressure indication is needed to feed the hydrogen plant, but should also be sent to the MCR for awareness by the operators.
- The steam inlet valve(s) are assumed to be AOVs that can modulate flow in the system and close on automatic signals as needed.
- The reboiler outlet valves can be manual valves for the purpose of maintenance/isolation.
- All items in the MCR can be left as analog controls or can be fed into a digital control system (DCS).
- The status of the hydrogen plant can be fed to a plant process computer or can be monitored remotely. It is only needed for business purposes (trending and monitoring) and is not critical to the operators in the MCR of the nuclear plant.

The above items, as described, are typical items for a nuclear facility. All items have been previously used in a nuclear facility. As such, it is expected that these items can be screened out of the 10 CFR 50.59 process and would not require a 10 CFR 50.59 evaluation. Because the steam is taken from the secondary side of the plant, all items are non-safety related. If digital controls are used, digital/cybersecurity process rules are not expected to challenge the results of the 10 CFR 50.59.

C-5. Regulatory Strategy Subcommittee

The Regulatory Strategy Subcommittee informed the AE design activities related to NRC regulatory requirements required prior to implementation of new plant features needed to implement HTE within the owner-controlled area of an existing NPP.

Any modification that would involve a need to revise the plant technical specifications (TS) must be submitted as a LAR for prior NRC approval. The criteria for determining the need for a TS change is specified in 10 CFR 50.36, “Technical Specifications.” Pending satisfaction of the four criteria within 10 CFR 50.36, an assessment is then required to determine if the associated change in the current licensing basis (CLB) meets the criteria specified in 10 CFR 50.59 for proceeding with the implementation of a plant modification without prior NRC approval.

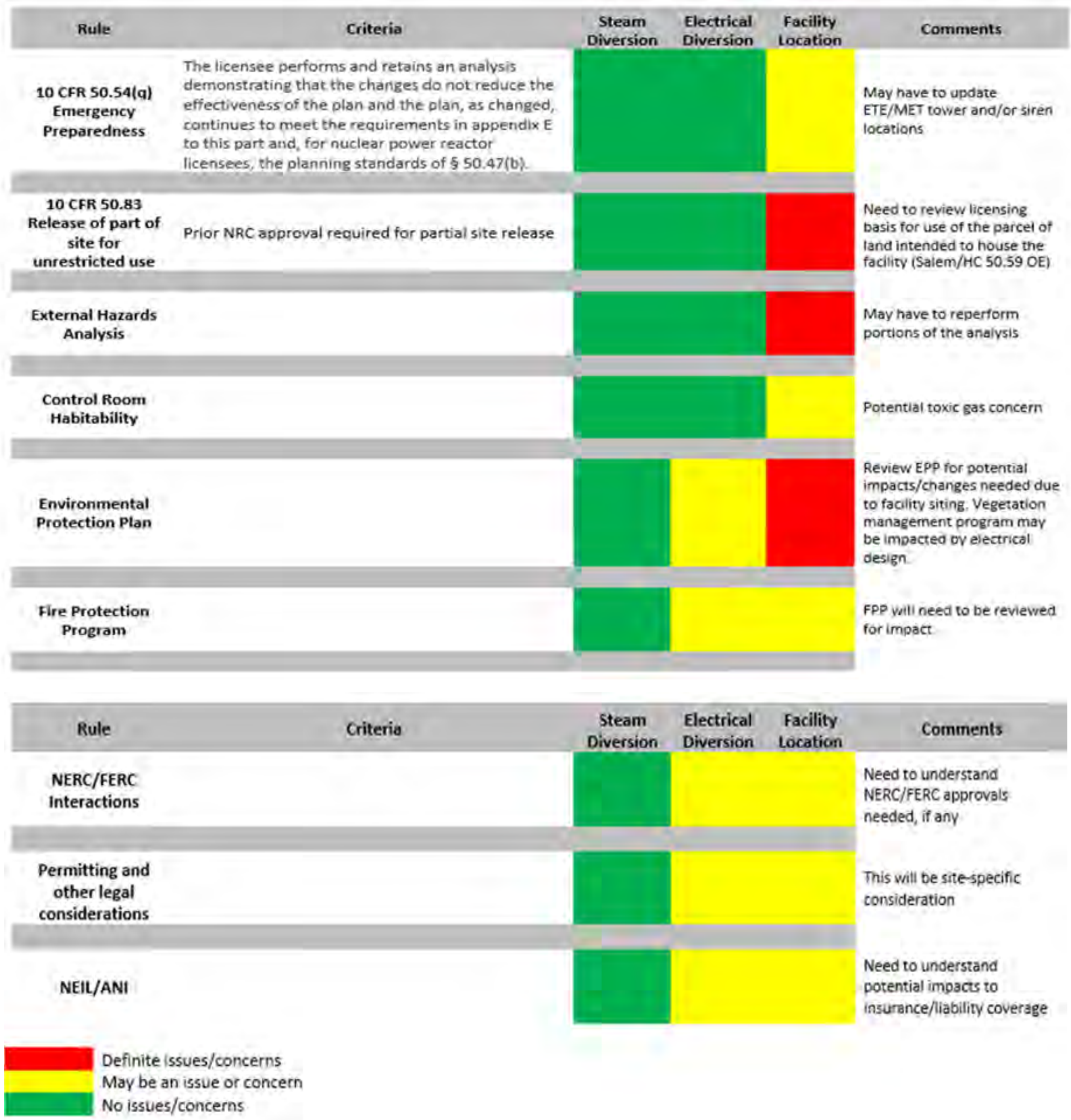
The primary focus of the Regulatory Subcommittee was to assess the existing AE design against the four criteria of 10 CFR 50.36 and the eight criteria specified in 10 CFR 50.59 as well as to inform the AE of additional NRC regulatory requirements that could, based on unique plant specific CLBs, drive the need for additional NRC interactions.

To accomplish the task, this team created a worksheet (below) that included the specific regulations and associated criteria necessary for success. For each criterion, an assessment was made associated with the HTE design aspects of steam diversion, electrical diversion, and facility location.

REGULATORY STRATEGY SUBCOMMITTEE WORKSHEET

Rule	Criteria	Steam Diversion	Electrical Diversion	Facility Location	Comments
10 CFR 50.59	Increase frequency of occurrence of an accident				Not a concern if steam tap is located downstream of turbine stop valves
	Increase likelihood of SSC malfunction				Electrical: need to ensure power is not diverted from safety buses Facility: Results of blast analysis may show potential to impact SSC
	Increase consequences of an accident				
	Increase consequences of SSC malfunction				
	Create accident of a different type				Dependent on external hazards already evaluated for site
	Create SSC malfunction with a different result				
	Exceed or alter design basis limit for fission product barrier				
	Depart from method of evaluation described in FSAR				MOE needs to be NRC approved
10 CFR 50.92 No Sig Hazard	Increase in probability or consequence of accident previously evaluated				Aligns with 50.59 criterion 1
	Create possibility of new or different kind of accident from any previously evaluated				
	Involve significant reduction in margin of safety				Aligns with 50.59 criterion 5 Tied to RG 1.91 blast analysis results

Rule	Criteria	Steam Diversion	Electrical Diversion	Facility Location	Comments
10 CFR 50.36 Technical Specifications	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.				
	A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.				Be aware of Salem/HC OE on facility siting in OCA/PA (Design Features section of TS)
	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.				
	A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.				
10 CFR 50.54(p) Security	The changes do not decrease the safeguards effectiveness of the plan.				• Steam/Electrical: penetrations of new system need to be reviewed/evaluated • Facility: RG 1.91 blast analysis will likely be needed • Potential new target set created • Cyber security considerations for new facility



Rule	Criteria	Steam Diversion	Electrical Diversion	Facility Location	Comments
10 CFR 50.54(q) Emergency Preparedness	The licensee performs and retains an analysis demonstrating that the changes do not reduce the effectiveness of the plan and the plan, as changed, continues to meet the requirements in appendix E to this part and, for nuclear power reactor licensees, the planning standards of § 50.47(b).				May have to update ETE/MET tower and/or siren locations
10 CFR 50.83 Release of part of site for unrestricted use	Prior NRC approval required for partial site release				Need to review licensing basis for use of the parcel of land intended to house the facility (Salem/HC 50.59 OE)
External Hazards Analysis					May have to reperform portions of the analysis
Control Room Habitability					Potential toxic gas concern
Environmental Protection Plan					Review EPP for potential impacts/changes needed due to facility siting. Vegetation management program may be impacted by electrical design.
Fire Protection Program					FPP will need to be reviewed for impact
NERC/FERC Interactions					Need to understand NERC/FERC approvals needed, if any
Permitting and other legal considerations					This will be site-specific consideration
NEIL/ANI					Need to understand potential impacts to insurance/liability coverage

Definite issues/concerns
May be an issue or concern
No issues/concerns

10 CFR 50.36: TS considerations regarding establishing limiting conditions for operation of a nuclear reactor for each item meeting one or more of the following criteria:

- Criterion 1. Installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary.
- Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
- Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
- Criterion 4. A structure, system, or component, which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The initial Regulatory Subcommittee assessment has identified aspects for further discussion relative to a potential TS revision relative to TS primarily involve the impact of drawing steam from the plant and the HTE equipment citing. A TS revision relative to the electrical design should not require a TS change, provided power remains ensured to the safety buses. The following is a summary of the areas identified to date for further evaluation:

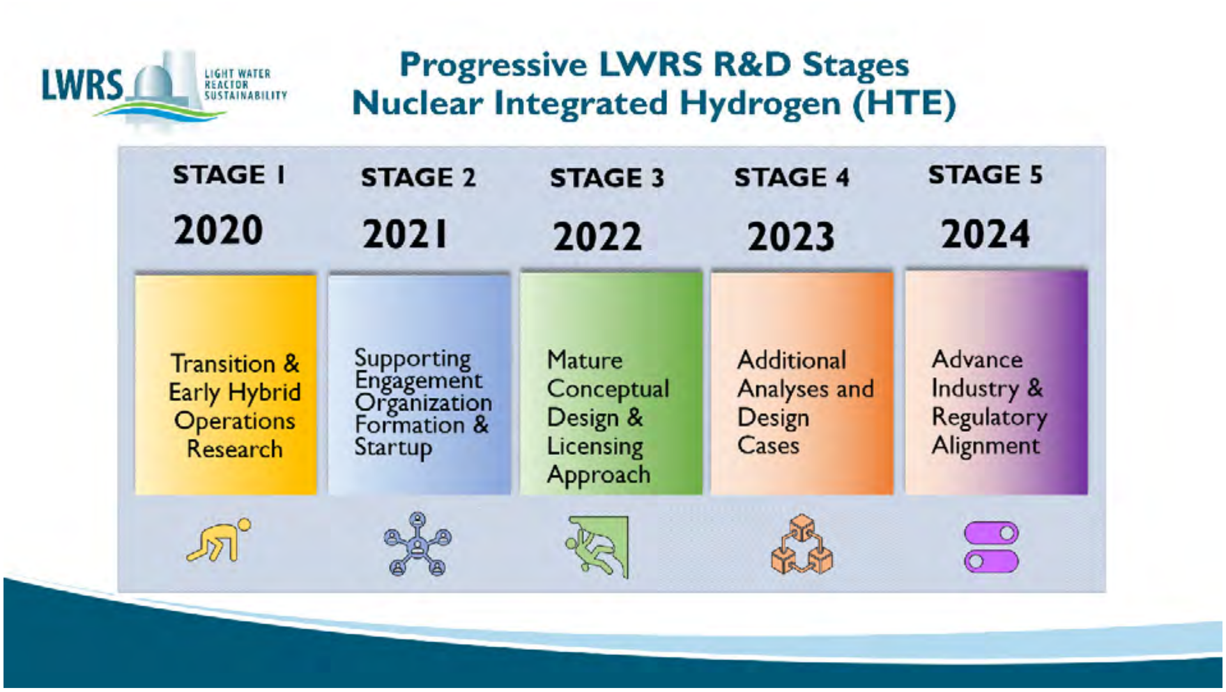
1. Should the HTE design for the associated steam and return isolation valves/devices and associate control logic necessitate credit for these devices to limit the frequency of a step load change event, then these valves may need to be included in TS to require a surveillance and limit their out of service duration.
2. The HTE equipment location is important to prevent a challenge to existing plant equipment and features necessary to mitigate a DBA or transient. This concern is relevant to Criterion 3 of 10 CFR 50.36.
3. A recent industry example where a map of the owner-controlled area is provided within the design feature section of the site-specific TS. The NRC recently cited this plant for plans to revise this map to reflect an alternate use of this property without prior NRC approval of a revision to the approved TS.

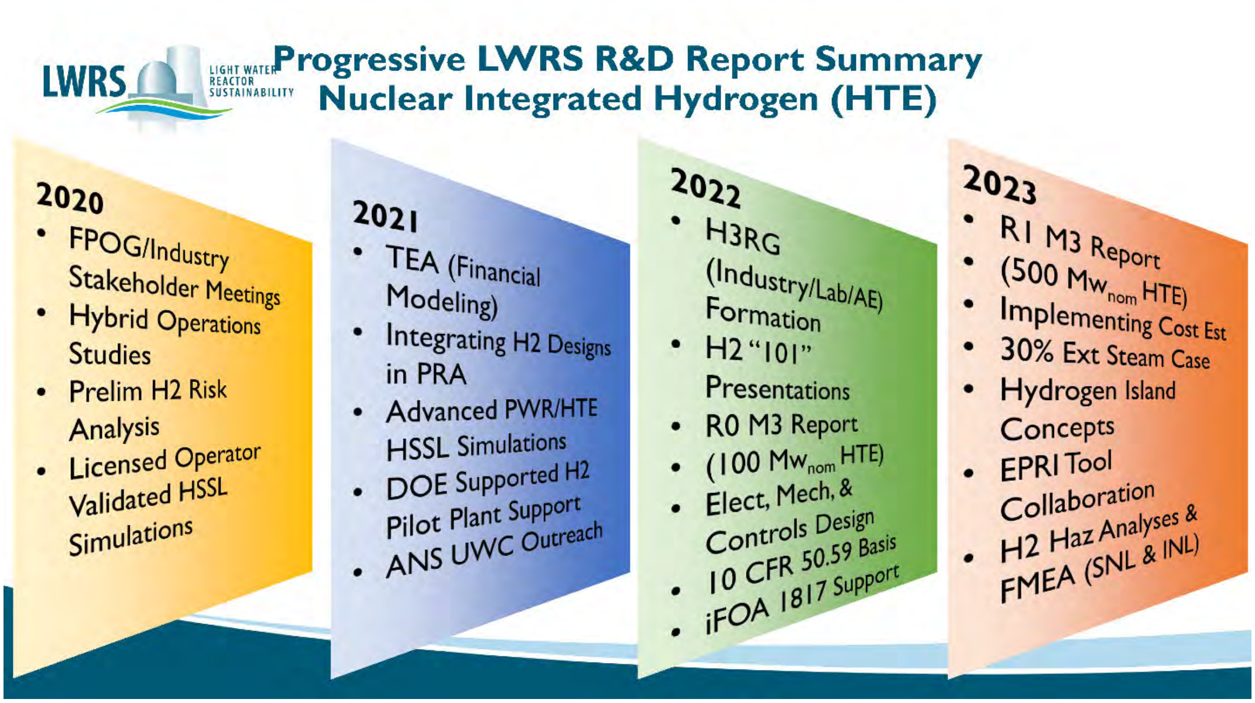
10 CFR 50.59: Changes, Tests, and Experiment Considerations:

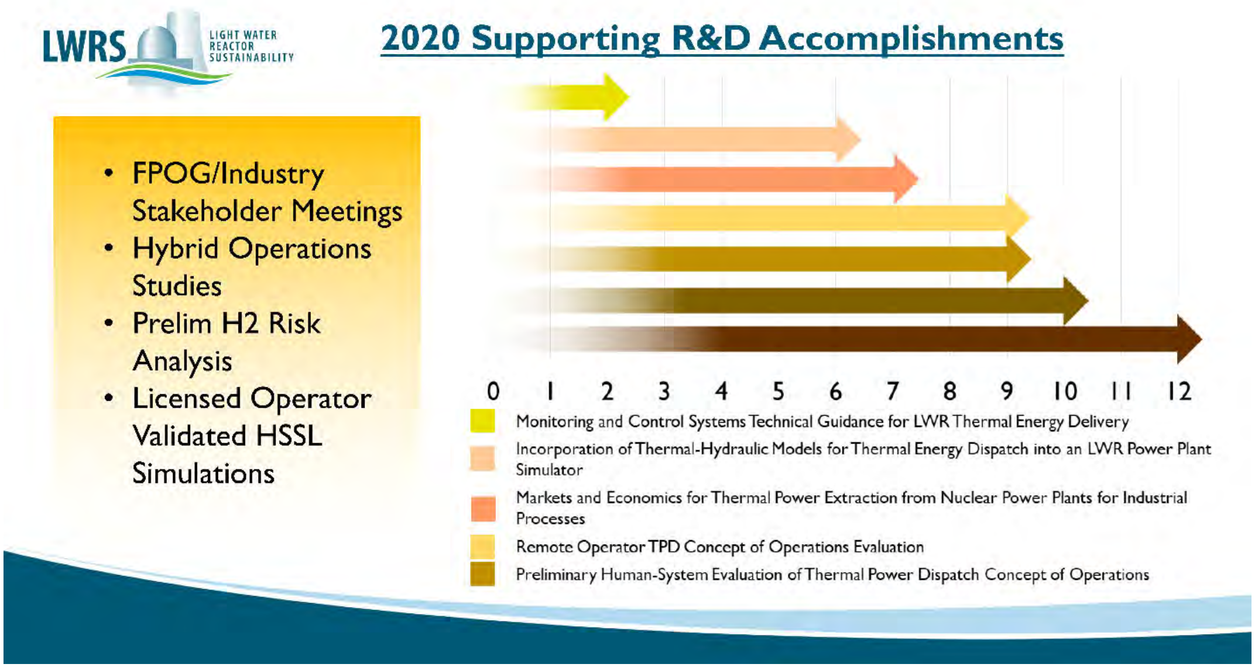
- Use of the 10 CFR 50.59 process is only allowed after determination that a change to the TS is not necessary.
- A 10 CFR 50.59 evaluation examines eight criteria for determination that a modification can be implemented without prior NRC approval.
- If any of the eight criteria are not met, then the 10 CFR 50.59 process cannot be used to implement the modification and an LAR must be submitted to the NRC for review and approval.
- The licensee is required to periodically submit to the NRC a list of all 10 CFR 50.59 evaluations that have been completed.
- The initial Regulatory Subcommittee assessment identified aspects for further discussion relative to a 10 CFR 50.59 evaluation, including the following:
    - A failure within the HTE associated steam and return isolation valves/devices and associate control logic must not result in more than a minimal increase in the frequency of occurrence of an accident.
    - A fault in the HTE associated electrical supply from the plant must not impact power to the safety buses and/or increase the frequency and likelihood of occurrence of an accident or SSC malfunction.
    - The addition of the hazards associated with the HTE modification must be shown to not impact existing plant SSCs. This aspect could be problematic for plants that do not currently have existing external hazards analyzed within their CLB.
    - The methods of analysis must be precisely aligned with the plant CLB or previously approved for the specific application. Assumptions used in the analysis must be clearly articulated and validated on a plant specific basis.

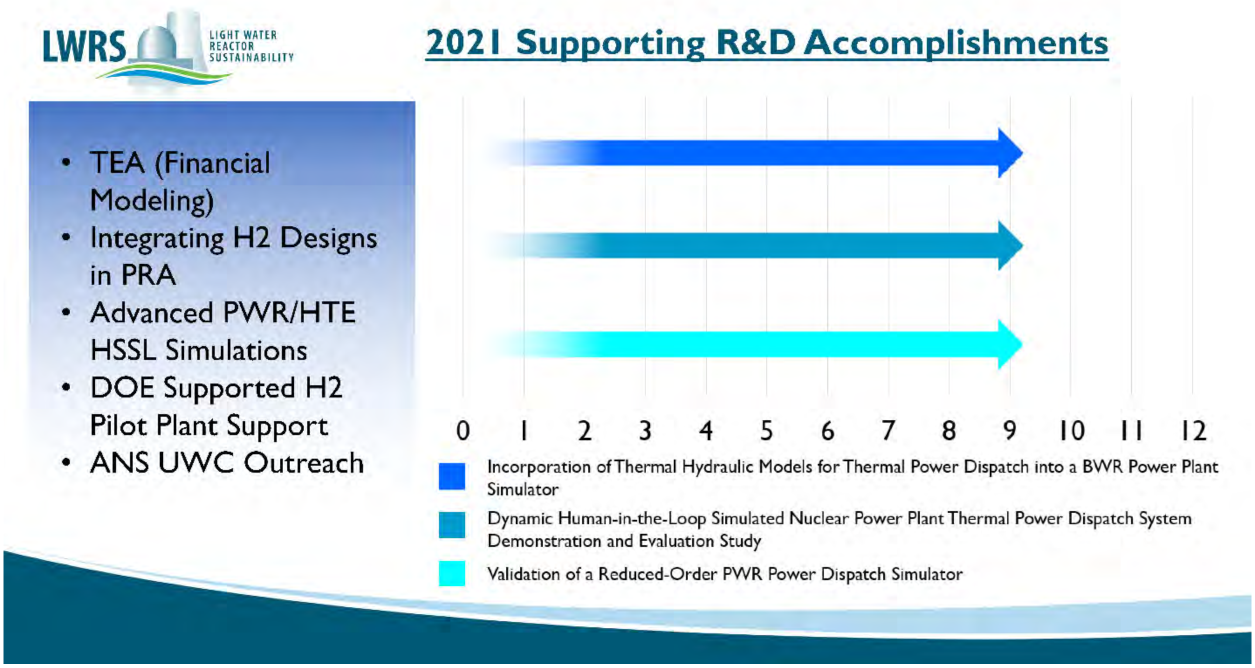
Appendix D

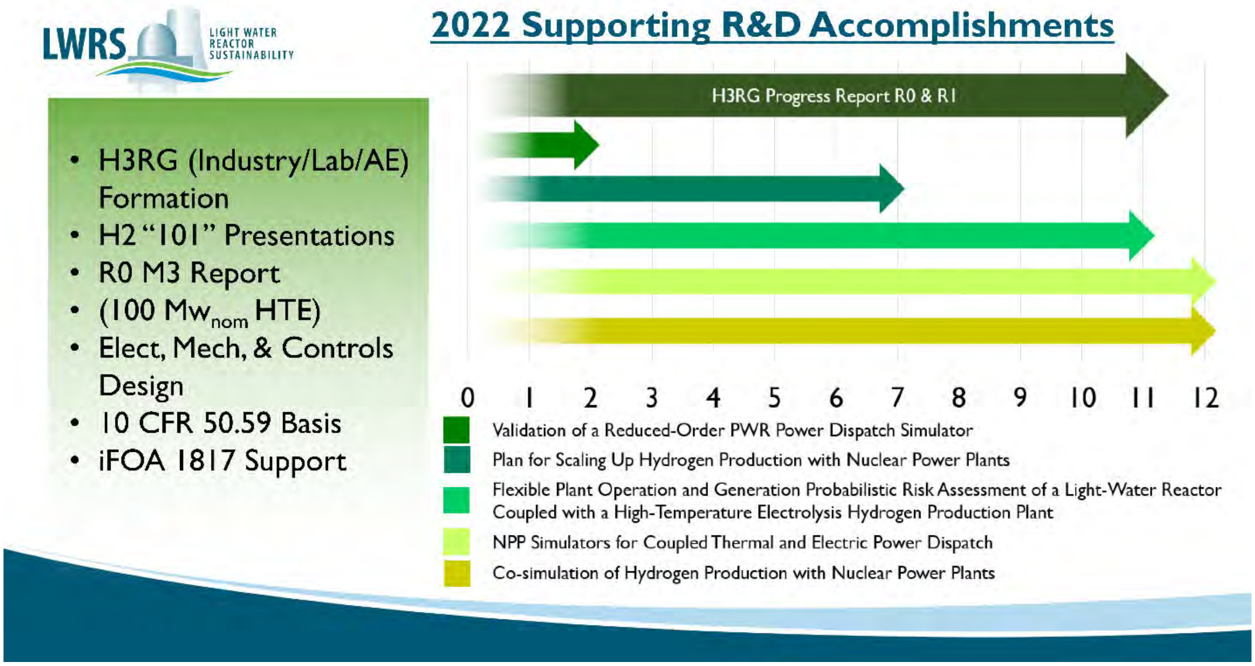
Supporting Research Deliverables

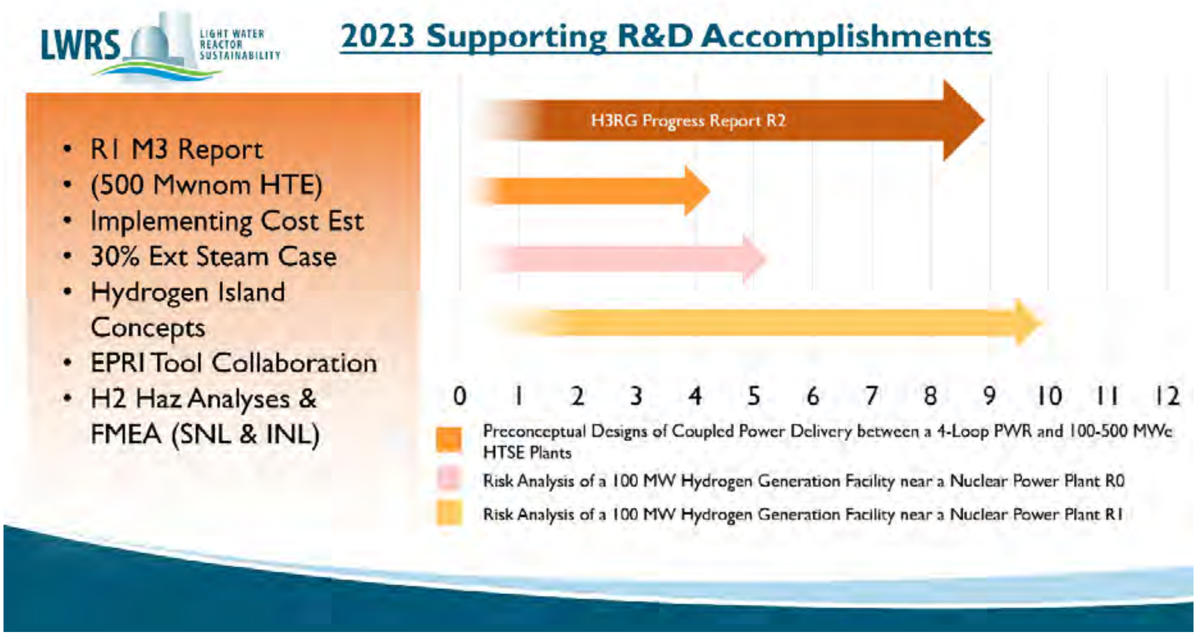





LWRS LIGHT WATER REACTOR SUSTAINABILITY
2020 Supporting R&D Accomplishments
• FPOG/Industry Stakeholder Meetings
• Hybrid Operations Studies
• Prelim H2 Risk Analysis
• Licensed Operator Validated HSSL Simulations
0 1 2 3 4 5 6 7 8 9 10 11 12
Monitoring and Control Systems Technical Guidance for LWR Thermal Energy Delivery
Incorporation of Thermal-Hydraulic Models for Thermal Energy Dispatch into an LWR Power Plant Simulator
Markets and Economics for Thermal Power Extraction from Nuclear Power Plants for Industrial Processes
Remote Operator TPD Concept of Operations Evaluation
Preliminary Human-System Evaluation of Thermal Power Dispatch Concept of Operations

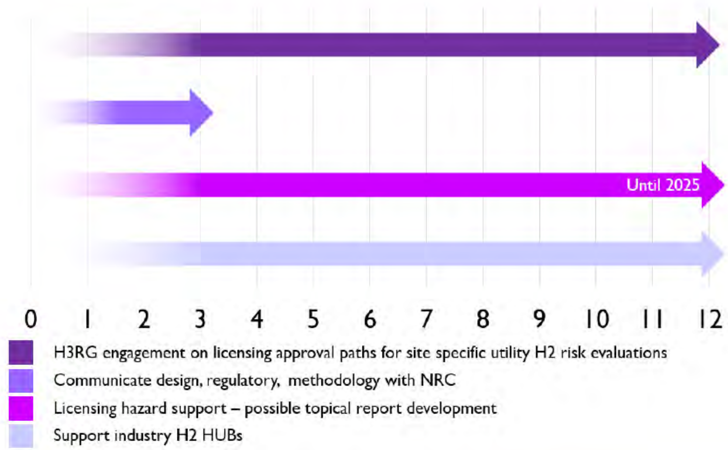

LWRS LIGHT WATER REACTOR SUSTAINABILITY
2021 Supporting R&D Accomplishments
• TEA (Financial Modeling)
• Integrating H2 Designs in PRA
• Advanced PWR/HTE HSSL Simulations
• DOE Supported H2 Pilot Plant Support
• ANS UWC Outreach
0 1 2 3 4 5 6 7 8 9 10 11 12
Incorporation of Thermal Hydraulic Models for Thermal Power Dispatch into a BWR Power Plant Simulator
Dynamic Human-in-the-Loop Simulated Nuclear Power Plant Thermal Power Dispatch System Demonstration and Evaluation Study
Validation of a Reduced-Order PWR Power Dispatch Simulator


LWRS LIGHT WATER REACTOR SUSTAINABILITY
2022 Supporting R&D Accomplishments
• H3RG (Industry/Lab/AE) Formation
• H2 "101" Presentations
• R0 M3 Report
• (100 Mw_{nom} HTE)
• Elect, Mech, & Controls Design
• 10 CFR 50.59 Basis
• iFOA 1817 Support
H3RG Progress Report R0 & R1
0 1 2 3 4 5 6 7 8 9 10 11 12
Validation of a Reduced-Order PWR Power Dispatch Simulator
Plan for Scaling Up Hydrogen Production with Nuclear Power Plants
Flexible Plant Operation and Generation Probabilistic Risk Assessment of a Light-Water Reactor Coupled with a High-Temperature Electrolysis Hydrogen Production Plant
NPP Simulators for Coupled Thermal and Electric Power Dispatch
Co-simulation of Hydrogen Production with Nuclear Power Plants


LWRS LIGHT WATER REACTOR SUSTAINABILITY
2023 Supporting R&D Accomplishments
• R1 M3 Report
• (500 Mwnom HTE)
• Implementing Cost Est
• 30% Ext Steam Case
• Hydrogen Island Concepts
• EPRI Tool Collaboration
• H2 Haz Analyses & FMEA (SNL & INL)
H3RG Progress Report R2
0 1 2 3 4 5 6 7 8 9 10 11 12
Preconceptual Designs of Coupled Power Delivery between a 4-Loop PWR and 100-500 MWe HTSE Plants
Risk Analysis of a 100 MW Hydrogen Generation Facility near a Nuclear Power Plant R0
Risk Analysis of a 100 MW Hydrogen Generation Facility near a Nuclear Power Plant R1

2024 Next Step R&D Goals

- H3RG engagement on licensing approval paths for site specific utility H2 risk evaluations
- Communicate design, regulatory, methodology with NRC
- Licensing hazard support – possible topical report development
- Support industry H2 HUBs



LWRS LIGHT WATER REACTOR SUSTAINABILITY

Sustaining National Nuclear Assets

http://lwrs.inl.gov

INL FPOG Report Link