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Revision 1

Light Water Reactor Sustainability Program

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

November 2022

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SUMMARY

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

Federal incentives and actions are aligning to expand the role of nuclear power as a viable and flexible contributor to the evolving national clean energy mix through programs and initiatives like nuclear 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 strategically explored by industries desiring to transition away from carbon-intensive energy operations.

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

- Alternate product stream market assurance to support decision-making for large capital modification investments
- Electric utility and public utility long-standing tradition bias for electric-only plant operating philosophies
- Design change complexity and regulatory uncertainty associated with modifications 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 with the strategic intent to levelize clean energy integration with intermittent wind and solar capacity. 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 integrated hydrogen production at NPPs is related to how supporting design changes would conform to the licensing regulatory framework required by the U.S. Nuclear Regulatory Commission (USNRC). Design changes are routinely performed at operating U.S. nuclear power reactors, in part through a process where the licensee confirms that the proposed design change is permitted under the Code of Federal Regulations (CFR), Title 10 Part 50.59. When a proposed change to the facility is determined not to be within the limits specified in 10 CFR 50.59, a formal license amendment request (LAR) and specific USNRC approval are required under 10 CFR 50.90.

The Hydrogen Regulatory Research Review Group (H3RG) was formed to begin identification of the generic technical and safety risks that could be accepted under a 10 CFR 50.59 evaluation and thus avoid the uncertainty of the LAR process. Thus, the generic guidance serves to reduce 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, and nuclear utility licensing and design experts. Revision 0 to this report 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. High-temperature steam electrolysis (HTE) was evaluated as the paired hydrogen-generating technology design case based on use of both clean electric and clean steam to achieve higher efficiencies than the electric-only limitations of low-temperature

electrolysis (LTE). The simplified conceptual design and research and development (R&D) work performed under Revision 0 to this report [1] demonstrated early supporting design and regulatory approval path elements.

Revision 1 of this report further evaluates and explains subsequent regulatory research findings with specific emphasis on the likely degree to which the simplified 10 CFR 50.59 utility self-approval process may be used. These regulatory R&D-based conclusions are provided in the form of a generic 10 CFR 50.59 evaluation.

The generic 50.59 evaluation presented here will help readers assess the general feasibility of a nuclear-integrated hydrogen facility addition like that described. However, the generic evaluation does not address the wide range of regulatory requirements and licensing bases for the existing NPP fleet. This means that the generic evaluation should be taken as indicative, not determinative, of results expected from a site-specific evaluation. Users will need to perform detailed site-specific 50.59 evaluations to determine conformance with their specific NPP's design and current licensing basis. H3RG review confirmed that unique plant-specific licensing aspects will need to be considered as large-scale demonstrations, and final commercial designs are considered by nuclear utilities. Although these plant and license specific considerations were not practical to evaluate under the scope of this generic design and operating license effort, the H3RG identified and documented several areas that should be considered as a future subject of site-specific regulatory review evaluations.

CONTENTS

FIGURES

TABLES

ACRONYMS

AE	architectural engineer
BIL	Bipartisan Infrastructure Act
BWR	boiling water reactor
CDF	core damage frequency
CFR	Code of Federal Regulations
CLB	current licensing basis
DBA	design basis accident
DOE	Department of Energy
FEED	front-end engineering and design
FMEA	failure modes & 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
HP	high pressure
HTE	high-temperature steam electrolysis
HTEF	high-temperature electrolysis facility (hydrogen island)
I&C	instrumentation and control
iFOA	Industrial Funding Opportunity Announcement
IIJA	Infrastructure Investment and Jobs Act
INL	Idaho National Laboratory
kW	kilowatt
LAR	License Amendment Request
LBE	licensing basis events
LCOE	levelized cost of energy
LCOH	levelized cost of hydrogen
LERF	large early release frequency
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 rating)
MW_{th}	megawatt thermal rating (thermal power)
NPP	nuclear power plant
P&ID	piping and instrumentation diagram
PA	protected area
PRA	probabilistic risk assessment
PWR	pressurized water reactor
R&D	research and development
RIPE	Risk Informed Process for Evaluation
S&L	Sargent & Lundy
SNL	Sandia National Laboratories
SSC	system structure or component
TS	technical specifications
UFSAR	updated final safety analysis report
USNRC	Nuclear Regulatory Commission

REPORT ON THE CREATION AND PROGRESS OF THE HYDROGEN REGULATORY RESEARCH AND 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 continues to contribute to the unproductive expansion of time-of-day excess clean generation. The overlapping impact of the dominant clean generating sources (intermittent renewables and baseload nuclear power) exacerbates this challenge during 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 no additional grid demand for clean energy. During these times, nuclear could flexibly produce real-time usable or storable clean energy to decarbonizing functions across the power, industrial, agricultural, and transportation sectors. Specifically, hydrogen by electrolysis as a flexible energy stream from the existing nuclear fleet 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 in clean hydrogen production and global decarbonization of transportation, industrial, and other sectors have increased. 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 likely be an important and diversified national complement to limited renewable electricity storage via Lithium-Ion batteries and other emerging storage technologies. High-temperature steam electrolysis (HTE) systems can achieve relatively 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 clean, high-efficiency hydrogen by HTE, shown in Figure 1.

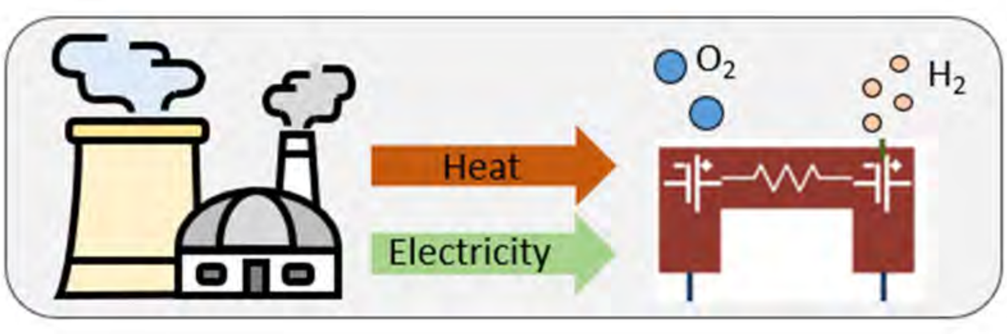

Figure 1. Nuclear provides heat and electricity for high-temperature electrolysis.

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 being addressed through exploration of practical pre-conceptual designs, pilot hydrogen projects, and development of likely licensing success paths consistent with the United States Nuclear Regulatory Commission (USNRC) requirements. This licensing R&D review element continues to be developed by nuclear industry design and regulatory experts under the H3RG.

A recent INL study [3] evaluated the feasibility of nuclear integrated HTE. A process design model was created that considered the performance of basic steady state, constant hydrogen production scenarios. The study evaluated the feasibility of HTE equipment utilizing the full energy output from a 1200 MWe light water reactor (LWR) to produce approximately 700 metric tons of hydrogen per day. This would require approximately 5% of total steam flow to provide process-heat input to vaporize process water feed stock to the electrolysis equipment. Most of the reactor-produced steam flow would continue to provide electrical generation both to meet HTE process power demands and to provide continued clean energy to the grid. Within these parameters, the report concluded that, assuming energy from the LWR for the HTE system is purchased at a price of $30/MWh, the base case HTE plant can produce pure hydrogen at a levelized cost of hydrogen (LCOH) of $1.90/kg, excluding product storage or transportation costs. The parameters that have the greatest impact on LCOH are energy price and the direct capital cost. Near-term technology, process, human performance, and governance changes are under investigation by NPPs as operating cost reduction strategies that target practical pathways to levelized cost of energy (LCOE) in the $21/MWh-e range.

These techno-economic research findings conservatively did not assign special market or decarbonization value to hydrogen produced by water electrolysis in three important ways:

1. The low- or no-contaminant value proposition of clean hydrogen produced by water electrolysis which can serve very high-purity applications without additional processing
2. The inherently low carbon footprint of <0.5 kg of CO_2 per kilogram of produced hydrogen, which is significantly less than the clean hydrogen standard of 2 kg of CO_2 per kilogram of hydrogen produced

   Note: Comparatively, the dominant hydrogen production method of steam methane reforming without carbon capture and sequestration produces between 7 to 9 kg of CO_2 for each kg of hydrogen (more commonly characterized in tons of CO_2 per ton of hydrogen). Additionally, the production constituents of natural gas, inherently produce contaminants that must be further processed for high-purity hydrogen applications.
3. Potential payment for grid services that can be rendered when the NPP and hydrogen plant are able to coordinate the dispatch of power between the electricity grid and the numerous power inverters within the large hydrogen generation plant

2. LABORATORY AND INDUSTRY REGULATORY COLLABORATION

2.1 Design and Regulatory Evaluation Approach

2.1.1 Demonstration Design

Revision 0 to this report [1] provides the pre-conceptual design elements necessary to successfully 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) design.

This work provides a basis for a scaled-up DOE nuclear integrated HTE collaboration pilot project (beyond current industry small-scale hydrogen demonstration projects underway at U.S. nuclear utilities). In comparison to these small-scale kW-level pilots, nuclear HTE at the 100 MW_{nom} (100 MW_e/25 MW_{th}) scale is effectively an order of magnitude larger. This introduces more complex plant equipment integration, operational interaction, and regulatory considerations for evaluation. The 100 MW_{nom} medium-scale pilot project evaluation addressed in [1] provided conceptual thermohydraulic, electrical, and controls integration design basis analyses. Summary design case assumptions are provided in Appendix 2 for information.

Figure 2 and Figure 3 below provide a graphic representation of the planned progression of nuclear integrated HTE demonstration projects from small- to medium-scale.

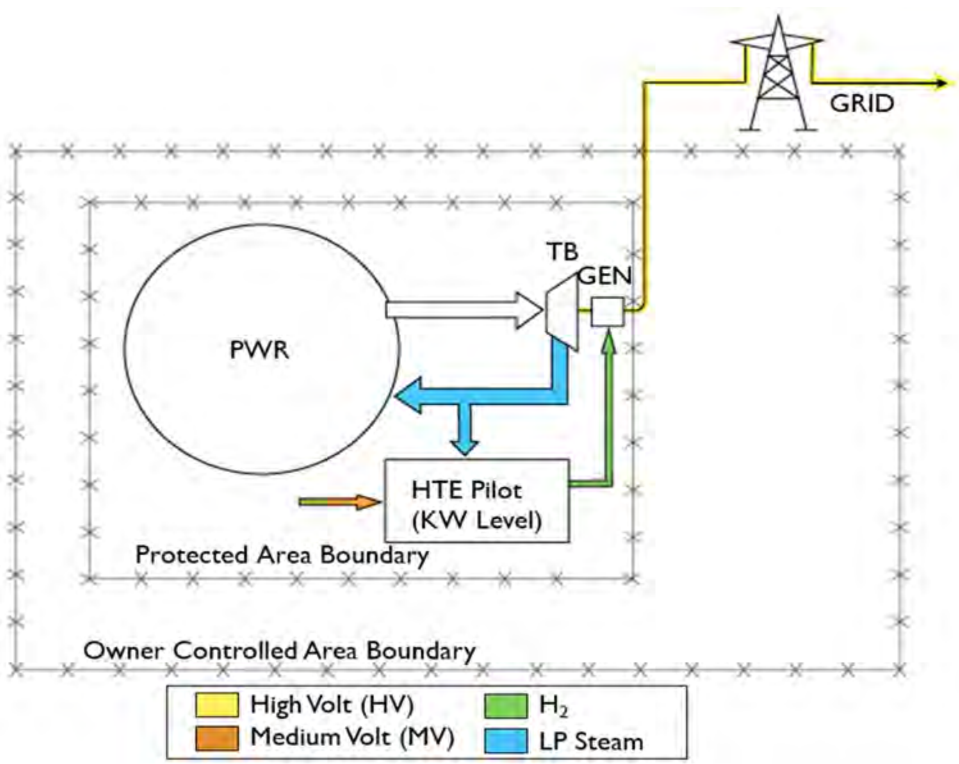

Figure 2. Early 2020s – Small kW-Scale HTE Demonstrations.

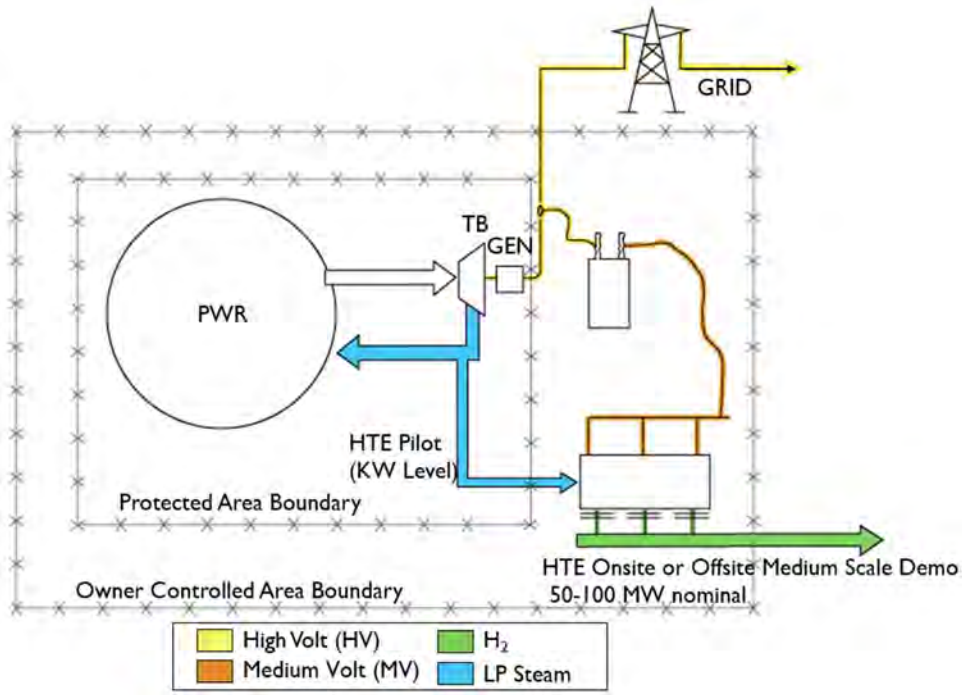

Figure 3. Late 2020s – Medium MW-Scale HTE Demonstrations

2.1.2 HTEF Linkage and Safety Analysis

All light water reactors employ on-site stored hydrogen gas facilities in support of plant processes, including main generator inerting requirements. Because nuclear integrated HTE introduces additional hydrogen gas volumes associated with HTEF gas production, distribution, compression, and storage functions, a generic probabilistic risk assessment (PRA) was performed [2] to quantify the impact of coupling two sizes of an HTEF (1150 MW_{nom} and 100MW_{nom}) with a NPP on design basis event 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, assumed failures were also evaluated for the capacity to increase 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 kilometer (km) from critical NPP components. This was based on sensitivity studies that were conducted to determine the minimum safe distance.

The modifications necessary to provide thermal energy from NPP steam transfer to a process steam supply for the HTEF introduces new equipment that must be evaluated for safety in the PRA. Two options were evaluated for effects on initiating event frequencies, CDF and LERF [2]. The largest initiating event increase caused by the addition of a heat extraction system (HES) was for a main steam line break (5.6%) which is considered minimal in plant modification licensing decisions.

The modifications necessary to provide direct electrical linkage from the NPP to the HTEF introduces new equipment that must be evaluated for safety in the PRA. A proposed design was evaluated for effects on initiating event frequencies, CDF and LERF [2].

The hydrogen detonation hazard analysis performed in [2] presents a strong safety case for locating the HTEF at 1 km from the NPP's switchyard transmission towers and safety-related equipment. The assumed bounding accident was from one of the electrolysis modules, therefore the size of the facility did not affect the overpressure potential, only the frequency that the event occurred. The bounding hydrogen detonation-caused initiating event was a switchyard-based LOOP (LOOP-SW). The large HTEF (1150 MW_{nom}) LOOP-SW initiating event frequency increased by 1.3% from nominal. The small HTEF (100 MW_{nom}) LOOP-SW initiating event frequency increased by 0.2% from nominal. While both increases are minimal, the increase in LOOP-SW frequency from the 100 MW_{nom} HTEF is relatively insignificant. The leak masses assumed in the PRA were very conservative (a 100% pipe rupture); hence, there is a likelihood that relocating the HTEF adjacent to the NPP would be safe and would adhere to regulatory requirements. Further, the report concluded in a sensitivity study that an HTEF could safely be located 0.5 km from these components [1&2]. This report and supporting generic 10 CFR 50.59 evaluation assumes HTEF location at 0.5 km, with acknowledgment that detailed siting at an actual NPP may be closer or more remote based on site-specific analysis. Such analysis is beyond the scope of this generic evaluation; however, for more information, [1] provides the typical guidance methodology to be employed for appropriate siting of an HTEF at an existing NPP.

2.1.3 Regulatory Considerations

The conceptual medium-scale 100 MW_{nom} design package issued under Reference [1] advances the path to nuclear integrated hydrogen by HTE at scale. A corresponding regulatory approval path must be identified for potential users to evaluate the licensing implications in such a design. This revision to Reference [1] explores such a pathway.

This medium-scale conceptual demonstration project involves design and systems integration aspects that have not specifically been evaluated previously for regulatory acceptance. This introduced the question whether the 10 CFR 50.59 evaluation process can be employed generically under the intent of that rule. Developing an understanding of the regulatory challenges to using 10 CFR 50.59 generically is an important project deliverable.

The provisions of 10 CFR 50.59 provide the principal criterion for changes at an NPP without additional USNRC review and approval (as well as for the performance of tests and experiments not described by the UFSAR). Use of this process is only allowed after determining that a change to the plant's Technical Specifications (TS) is not necessary. A 10 CFR 50.59 evaluation examines the following eight criteria describing a change to the facility for determination that a modification can be implemented without prior USNRC approval:

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 structure, system, or component (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 FSAR (as updated) being exceeded or altered
8. Does not result in a departure from a method of evaluation described in the FSAR (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 USNRC for review and approval. The licensee is required to periodically submit to the USNRC a list of all 10 CFR 50.59 evaluations that have been completed.

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

With that intent in mind, the H3RG was formed based on input from INL-led FPOG stakeholder meetings as shown in Figure 4, under several industry subcommittee research areas with leads selected from expert participants. These leads oversee reviews of individual regulatory subcommittee areas which provide design, operational, and regulatory strategy input. Also as shown in Figure 4 the regulatory input path to the current draft 10 CFR 50.59 (Appendix 1) has been informed by the collaborative efforts of INL researchers and contracted AE staff. This targeted use of expert review resources allowed for early identification of discrete subcommittee research consideration areas that have informed design and regulatory R&D efforts.

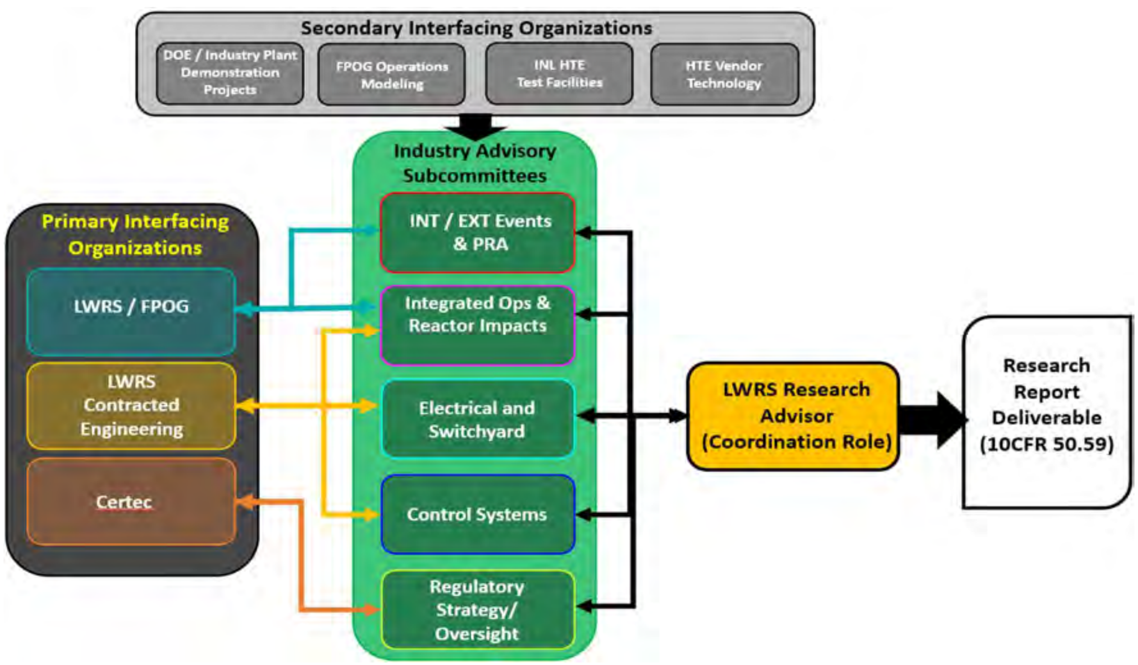

Figure 4. H3RG organizational and subcommittee structure.

During the H3RG formation stage, an experienced INL Program Advisor 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 USNRC LARs. This led to the establishment of foundational design and licensing considerations for laboratory and industry (H3RG) design consideration and inclusion in 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 stand-off requirements and on-site 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 assumption that integrated HTE design would be under site Operations control to ensure that system operation would have no unintended impacts on nuclear reactor power not directly under operator control and outside the requirements of 10 CFR 50.54. 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 operational business model.

Another formational 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 integrated licensing bases, and that a formal 10 CFR 50.59 evaluation will likely be required. In support of informing regulatory approval approaches under the 10 CFR 50.59 process, expert review also will be leveraged for:

- Comparative reviews of historical industry examples where approval of changes to the facility were appropriately completed under 10 CFR 50.59 – especially for first-of-a-kind and fundamental operating approach changes.
- Detailed reviews of historical 10 CFR 50.59 USNRC industry feedback 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.

A laboratory-contracted AE, Sargent & Lundy (S&L), brought extensive expertise in complex design and regulatory evaluation which significantly contributed to the collaboration. CERTREC Corporation was also contracted to manage the organization and work coordination of the H3RG industry group based on extensive nuclear experience with facilitation of complex regulatory interactions with the USNRC.

Based on these collaborative efforts, this report revision specifically describes findings to date that are related to regulatory approval feasibility via a non-plant-specific evaluation under the utility self-directed 10 CFR 50.59 evaluation process.

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].

This report summarizes the high-level background, bases, and findings associated with a draft version of the 10 CFR 50.59 evaluation for the proposed 100 MW_{nom} HTE demonstration design. Appendix 1 and summary Section 2.2.1 to this report provide:

- A draft 10 CFR 50.59 deliverable for next-level review and comment by the H3RG team to assess the feasibility of the regulatory evaluation path to integration of a non-plant-specific medium-scale 100 MW_{nom} HTE capability. This is an important enabler to scaling up DOE-supported HTE demonstrations beyond the small-scale kW-level nuclear integrated HTE collaboration pilot projects currently underway at several U.S. nuclear sites.
- Early "site-specific" regulatory consideration areas which may represent typical U.S. nuclear fleet license condition challenge areas.

  Note that site-specific considerations identified during H3RG reviews have not been evaluated in detail as part of the scope of this draft 10 CFR 50.59 evaluation development process. This was determined to be an issue of practicality. As H3RG review progressed during the design review phase, it was recognized that unique site circumstances and the lack of a standard set of licensing basis elements across the U.S. nuclear fleet prohibited practical evaluation of an enveloping set of licensing requirements to include in a draft 10 CFR 50.59 product. It was thus determined that typical site-specific areas would be identified and captured for future use by nuclear utility demonstration or full-scale implementation designers and regulatory staff to help build FEED study considerations to help characterize the scope of site-specific licensing requirements to be addressed. Site-specific review considerations identified during the review are documented in Reference [1] and the CERTREC web-based drive work platform used by the H3RG. Section 2.2.2 makes appropriate section ties to the individual subcommittee considerations that may be of value to future FEED study developers. These are not intended to be all-inclusive of the site-specific review areas that must be considered.

Where plant specific licensing evaluations conclude that prior regulatory approval is required, utilities who qualify for the use of the Risk Informed Process for Evaluation (RIPE) may perform requisite probabilistic analyses to quantify the risk significance of hydrogen hazards. "Site-specific" probabilistic hazard analyses that demonstrate that the changes are of low safety significance are expected to be able to request expedited NRC LAR approval. Alternatively, the utility could use the standard 10 CFR 50.90 license amendment process.

Although the intent of ongoing regulatory research is to provide a utility basis for nuclear integration of HTE within the bounds of 10 CFR 50.59, other regulatory approval paths may need to be addressed (especially as the size of HTE projects increase). These could fall within two areas:

1. Large-Scale Integration Complexities

   To date, no conflicting regulatory areas have been identified on the medium-scale conceptual generic design for the 100 MW_{nom} HTE demonstration project that would preclude the use of the 10 CFR 50.59 process. As the next phase of the design project progresses to a larger scale 500MW_{nom} demonstration plant design, there is the potential to identify "dividing lines" between the use of 10 CFR 50.59

evaluation and LAR processes. Although not indicated in the draft work completed so far, if research points to specific design or operational issues that fall outside successful evaluation under 10 CFR 50.59, those areas will be identified and considered for additional research that could subsequently justify the plant changes under 10 CFR 50.59. Any design or licensing consideration areas that have been selectively screened out of this initial research project scope will be documented for future R&D consideration.

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 USNRC. The current design and regulatory evaluation work in support of the medium-scale 100 MW_{nom} (and next-generation 500 MW_{nom} demonstration project underway) cannot address the full spectrum of such plant-specific licensing nuances. It is expected that any applicable plant-specific considerations requiring USNRC approval will emerge as utility candidates for medium-scale HTE adoption are selected through DOE sponsored activities like nuclear hydrogen hubs and Industrial Funding Opportunity Announcement (iFOA) awards or during utility design change development outside those award processes. Additional H3RG generic guidance may be developed to support utilities that cannot successfully justify plant modifications under 10CFR50.59 but could seek to apply the RIPE approach for which engineering, and operations measures can ensure that the probabilistic analyses will demonstrate that the changes are of low safety significance.

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 adoption feasibility aspects. This has not yet been determined but would support moving forward on nuclear integrated HTE adoption while minimizing potentially repetitive utility-required design and regulatory support products. This may be particularly useful for plant-specific project commonalities that may not support regulatory justification under a 10 CRF 50.59 evaluation.

Even where USNRC approval may be required, future laboratory research deliverables could be developed to provide a basis for meeting the intent of 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 whose PA boundary is located closer to safety-related SSCs than the 500-meter stand-off assumptions used in the current demonstration design basis.

2.2 Summary Regulatory Findings and Considerations

2.2.1 Draft 10 CFR 50.59 Summary

The first phase of the lab-contracted 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 as 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 were reviewed and discussed with the H3RG. This work pointed to a favorable outcome from utilization of 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, which is included within Appendix 1. This generic evaluation assesses all eight criteria of 10 CFR 50.59 (c)(2) for the acceptability of implementing the Reference [1] pre-conceptual design which integrates a medium-scale 100 MW_{nom} HTE capability as a proposed change to an assumed 1200 MWe GPWR. This evaluation also represents a generic evaluation guide for use by industry stakeholders considering the proposed coupling of a NPP to a hydrogen production facility.

Although it is likely that the findings of this draft 10 CFR 50.59 evaluation will have strong conclusion applicability to plant electrical, thermohydraulic, and controls plant integration, as described above, it does not necessarily represent the full evaluation context needed to address all "site-specific" licensing requirements that may be needed for USNRC approval and/or additional research. Thus, it is expected that some nuclear facilities will likely have specific written considerations in their licenses that are more difficult to address than others. In other words, a generic 10 CFR 50.59 document will not be a one size fits all solution but is expected to be of use by industry with varying degrees of customization with additional consideration of site-specific formatting requirements.

Table 1 summarizes the eight 10 CFR 50.59 criteria and high-level summary descriptions and how each is met as described in detail in Appendix 1.

Table 1. 50.59 Summary Response Bases.

Criteria	50.59 Question Summary	Summary Response Bases
1	More than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?	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. A change from the max to the min (or from the min to the max) 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. 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.
1		Sudden changes in the electric power supplied to the HTEF would cause the balance between the electric power provided by the main generator to the HTEF and to the switchyard and transmission system to be readjusted, with the main generator and switchyard remaining stable.

Table 1. 50.59 Summary Response Bases (cont.)

Criteria	50.59 Question Summary	Summary Response Bases
1		<u>Loss of Offsite Power</u> 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.
1		<u>High Energy Line Break</u> Routing of steam in the new piping (10” maximum diameter) from the cold reheat piping to the new reboiler 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. Small addition of new piping and components in the plant and will be designed to the same codes and standards. No more than a minimal increase in the frequency of occurrence of a high energy line break. Represents a small addition to the amount of high energy piping.
1		<u>Flooding</u> New piping and components represent a small addition to the existing piping and components which have the potential to initiate a flooding event. New piping, tank, and components will be designed to the same codes and standards. Likelihood of a failure in the new piping and components remains very low.
1		<u>Turbine-Generator Trip</u> Installation of new electrical devices at the high-voltage side of the main transformer 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. Designed to the codes and standards appropriate for this application, such that these types of faults or failures are exceedingly rare.

Table 1. 50.59 Summary Response Bases (cont.)

Criteria	50.59 Question Summary	Summary Response Bases
2	More than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the FSAR?	New Equipment Impacts on SSC Malfunctions 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) Evaluations performed for the siting of a generic HTEF determined that the HTEF electrolyzers should be located at least 500 meters from SSCs important to safety [Reference1]. This minimum distance represents the point at which the overpressure effect from the maximum credible accident at the HTEF would fall below 1 psi, thus preventing a malfunction of a SSC important to safety.
3	More than a minimal increase in the consequences of an accident previously evaluated in the FSAR?	The existing plant systems include steam piping and associated components whose failure could have radiological consequences. The bounding analysis for new equipment failure events is the analysis of the radiological consequences of the rupture of a main steam line. The analysis remains bounding for the case of a rupture in the much smaller steam lines to be installed under the proposed activity.
4	More than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR?	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. 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.

Table 1. 50.59 Summary Response Bases (cont.)

Criteria	50.59 Question Summary	Summary Response Bases
5	Possibility for an accident of a different type than any previously evaluated in the FSAR?	The HTEF, which will be producing hydrogen nearby, introduces the potential for an explosion or fire at that facility to affect SSCs on site. Evaluations performed for the siting of a generic HTEF determined that the HTEF electrolyzers should be located at least 500 m from SSCs important to safety [References 1&2]. This minimum distance represents the point at which the overpressure effect from the maximum credible accident at the HTEF would fall below 1 psi. An evaluation of the explosion hazard from the proposed HTEF has not been 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. Operator errors involving the new operator controls for steam and electric power to the HTEF could initiate, at most, slight transients in balance-of-plant systems. Question 1, concluded such transients do not reach the threshold of an "accident previously evaluated in the FSAR."
6	Possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR?	The installation of new steam, condensate, and demineralized water piping and components, and the installation of new high-voltage electrical equipment that introduce the potential for a catastrophic failure in any of these to affect SSCs important to safety. 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. 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.

Table 1. 50.59 Summary Response Bases (cont.)

Criteria	50.59 Question Summary	Summary Response Bases
7	Design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?	The installation of new mechanical and electrical equipment that introduces the potential for failures of these new SSCs to produce minor changes in the flow of steam, condensate, demineralized water, or electrical power. Such changes are not associated with any fission product barrier.
8	Departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?	New transient and equipment failure effects - evaluation on methods described in the FSAR, design bases, or safety analyses. The methods of evaluation used in determining the effects of a transient in the cold reheat or other extraction steam systems and in determining the effects of a transient in the plant electrical systems are not described in the UFSAR for the reference plant. No methods of evaluation for determining the impacts of a H_2 fire or explosion on site are described in the FSAR for the current small H_2 quantities used. It is expected that an evaluation of the explosion hazard from the proposed HTEF (and its larger H2 quantities) by the relevant Regulatory Guide 1.91 criteria for the distance from the HTEF to SSCs important to safety would be met. Such an evaluation would use the same method of evaluation used in Regulatory Guide 1.91 and be approved by the USNRC for the intended application.

2.2.2 Subcommittee-Specific Considerations

Appendix 3 includes noteworthy H3RG working subcommittee excerpts regarding design assumptions, references, and regulatory considerations that should be comparatively considered in next-step plant-specific FEED study activities. In addition to evaluation areas described in Appendix 3, H3RG Regulatory Subcommittee assessment also identified subtle “siting” impacts that could require a TS revision including (but not limited to):

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 example where a map of the owner-controlled area is provided within the design feature section of the site-specific TS. The USNRC recently cited this plant for plans to revise this map to reflect an alternate use of this property without a revision to the approved TS with prior USNRC approval. This ongoing discussion is related to 10 CFR 50.83 requirements regarding partial release of the NPP site for unrestricted or alternate uses.

These type of TS issues (and others) are expected to be highly plant-specific based on the content of individual NPP TS’s (including less prominent site layout related information that might be contained in the Administrative Section of NPP TS). Additionally, introducing nuclear integrated HTE on a NPP site may be viewed as effectively changing site use intent as contained or inferred in the TS and will thus likely require certain USNRC notification and reporting requirements in addition to typical reporting of 50.71(e) UFSAR and 50.59 updates that are sent to the USNRC.

It is also expected that additional site-specific licensing-related evaluations will be required, including 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.

It has not yet been determined whether additional generic H3RG work will be provided to address a more comprehensive list of generic site-specific considerations since "site-specific" reviews for near-term iFOA award plants performing FEED studies will be forthcoming.

2.3 Summary Conclusions and Next Actions

2.3.1 Summary Conclusions

The 10 CFR 50.59 completed draft deliverable contained in Appendix 1 successfully assessed all eight criteria of 10 CFR 50.59 (c)(2) to demonstrate the viability of linking the Reference [1] issued conceptual 100 MW_{nom} nuclear integrated hydrogen HTE design with a standard 1200 MW generic PWR plant design. 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 projects. The selected 100 MW_{nom} conceptual design basis was provided to support expected near-term DOE and utility collaborations under the iFOA-1817 application process for hydrogen demonstration of nuclear integrated HTE at the MW-level.

No 50.59 question conflicts were identified with the early base design integration assumptions and pre-conceptual work done by the lab-contracted AE or based on review input provided by H3RG members. Ongoing laboratory testing and analytical R&D efforts continue to provide favorable results that, although not direct contributors to the 50.59 evaluation deliverables, are considered supportive findings in related areas such as equipment reliability, operational stability, and probabilistic design support.

Pending final review and input from the H3RG, this draft 50.59 evaluation represents a reasonable starting opportunity as a generic 50.59 evaluation template for use by industry parties considering the proposed coupling of a NPP to a hydrogen production facility. It is recognized that the findings of this draft 10 CFR 50.59 evaluation have the strongest conclusion applicability regarding plant electrical, thermohydraulic, and controls integration based on the limitation of pairing with the GPWR design. Conversely, it is likely that challenges to working under the 10 CFR 50.59 evaluation process will be revealed for some plants where GPWR modeling assumptions do not represent the full evaluation context needed to address certain "site-specific" licensing requirements. In these cases, license amendments may be required and/or additional research provided to mitigate these challenges. Thus, this emerging 10 CFR 50.59 template format is expected to be able to be used by 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.

2.3.2 Next Actions

The following directly related near-term nuclear integrated HTE research actions are committed/under consideration:

1. A 500 MW_{nom} plant conceptual HTE design is nearing draft issuance as a full-scale demonstration model and is scheduled for final issuance in the fourth quarter of FY-22.

   A R2 draft 10 CFR 50.59 evaluation for the issued conceptual 500 MW_{nom} HTE design will be reviewed by the H3RG as a separate licensing research deliverable and is scheduled for final issuance in the first quarter of FY23.

2. An integrated HTEF site linkage hazard analysis and typical low-pressure HTE vendor-specific equipment Failure Modes & Effects Analysis (FMEA) is being developed as a collaborative second quarter FY-23 research deliverable by INL & SNL in support of the generic pre-conceptual 500 MW_{nom} HTE design and associated 10 CFR 50.59 evaluation described in Item 1 above.

3. As expected iFOA 1817 DOE cost-share awards are issued to HTE demonstration plant candidates in 2023, one or more medium-scale projects are likely to start plant-specific FEED study evaluations for technical, economic, and licensing viability. This utility and AE work is expected to leverage research input from:

   - Completed 100 MW_{nom} and planned 500 MW_{nom} nuclear integrated design reports as described in Reference [1] and Item 1 above respectively
   - Completed Reference [2] probabilistic HTEF hazard and safety analysis and planned 2023 probabilistic research based on integrated HTEF site linkage hazards with typical low-pressure HTE vendor specific FMEA considerations (per Item 2 above)
   - Completed 100 MW_{nom} and planned 500 MW_{nom} generic 10 CFR 50.59 evaluation research reports as referenced and described herein.

   Lab-supported AE S&L, H3RG, and CERTREC opportunities to support FEED study development have not specifically been identified but is possible throughout 2023 as requested.

3. REFERENCES

1. Remer, S. J., R. Boardman, T. Westover, K. Vedros, T. Ulrich, J. Cadogan. 2022. “Report on the Creation and Progress of the Hydrogen Research Regulatory Review Group.” INL/RPT-22-66844, Idaho National Laboratory.
2. Vedros, K. G., R. Christian, and C. Otani. 2022. “Probabilistic Risk Assessment of a Light Water Reactor Coupled with a High-Temperature Electrolysis Hydrogen Production Plant.” INL/EXT 20 60104 Rev. 01, Idaho National Laboratory.
3. Wendt, D. S. and L. T. Knighton. 2021. “High Temperature Steam Electrolysis Process Performance and Cost Estimates.” INL/RPT-22-66117, Idaho National Laboratory.

Appendix 1
SL-017337 - “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 Hydrogen Production Facility

Report SL-017337

Revision 1

November 15, 2022

Project No.: A14248.006

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 hydrogen plants 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.

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 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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REVISION HISTORY

Revision	Issue Date	Notes
0	11/2/2022	• Background, Introduction, and 100 MW 50.59 evaluation developed from SL-016181, and updated based on comments from INL and H3RG (see Appendix A)
1	11/15/2022	• Administrative Update

TABLE OF CONTENTS

EXECUTIVE SUMMARY

Nuclear power has been identified as 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 100 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 100 MW_{nom} 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 100 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 (NPP) 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 NPP 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.

NPPs are typically operated at full (100%) power to provide baseload electrical power to the national grid. In deregulated markets, NPPs face economic pressures from fluctuating electrical demand and decreasing prices of wind and solar. Exploring alternative uses for the clean steam produced by NPPs during these challenging times is critical to improving the viability of 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 an NPP 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 industry 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 NPP. 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 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 100 MW_{nom}. The parameters describing this facility are shown in Table 1 below.

Table 1. 100 MW_{nom} Hydrogen Production Facility Pre-Conceptual Design Parameters

Parameter	Unit	Quantity
Hydrogen Production	U.S. tons/day	60
H_2 Plant Electric Load	MW_e	105
Total Electrical Power Requirements	MVA	140
H_2 Plant Thermal Load	MW_t	20
Total Thermal Power Requirements	MW_t	25

1.2. Reference Nuclear Power Plant

Both thermal and electrical power are required for operation; this power is supplied by a nearby NPP. 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 NPP'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 NPP.

1.4. Plant Interfacing

New piping connected to the exhaust of the main turbine (cold reheat) will provide steam to the plant secondary side of a new heat exchanger (steam reboiler), which will then supply steam to the HTEF in a tertiary loop. The condensate that forms on the plant secondary side of the reboiler will be returned to the main condenser. Demineralized water in the tertiary loop will be provided from a 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, a transformer to step down the voltage for use by the

HTEF, and medium-voltage cables (or buses) to connect to switchgear at the HTEF. This new electrical equipment will be located onsite.

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.

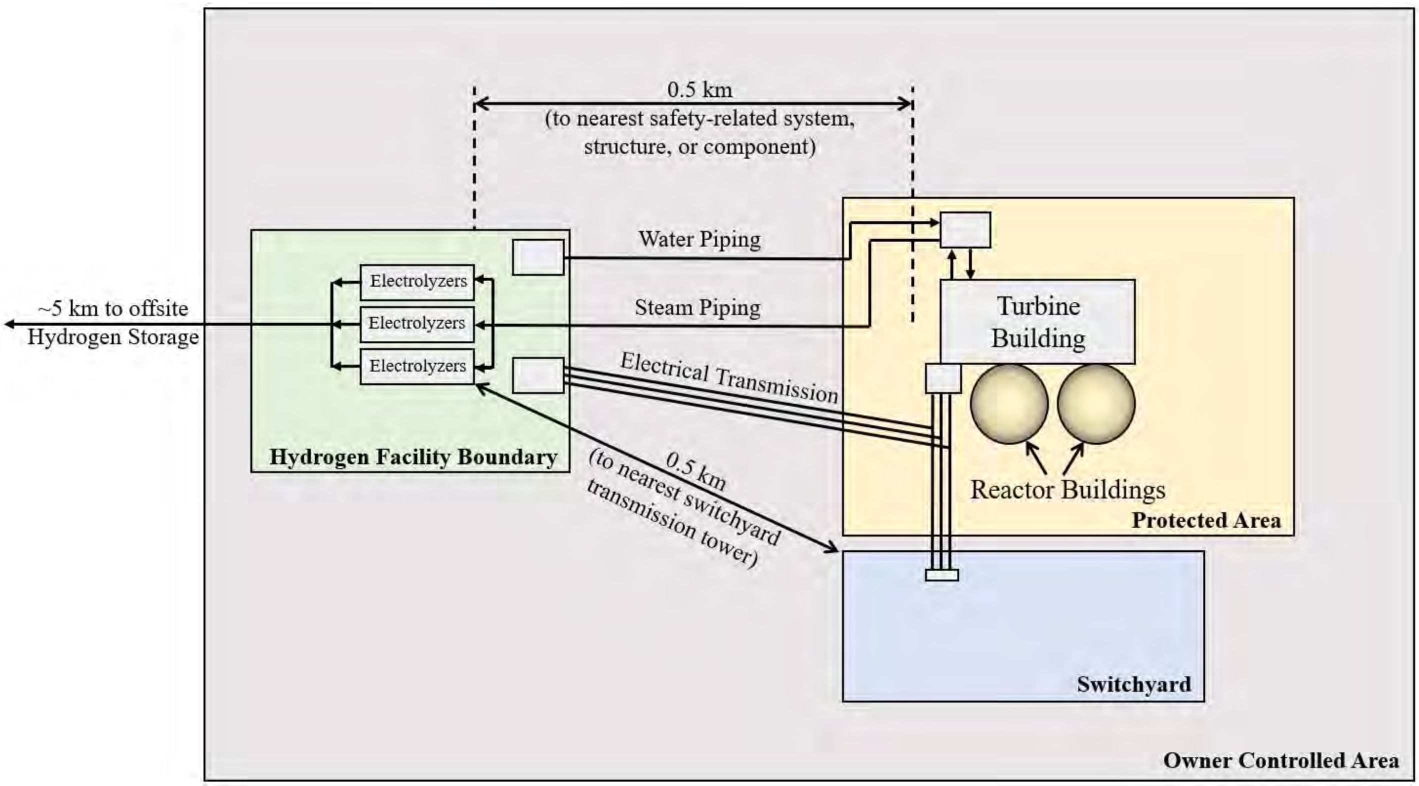


Figure 1. General Site Layout for Pre-Conceptual 100 MW_{nom} Hydrogen Facility Design

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 license amendment request 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 license amendment request 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 evaluations 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

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 [Reference 3] and cannot be used as a template for other NPPs. 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. Finally, it is noted that plants without an existing evaluation for explosive hazards in the vicinity (i.e., explosions at nearby facilities or on nearby transportation routes) may require a license amendment request.

3. 10 CFR 50.59 EVALUATION FOR 100 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 to 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.

The accidents previously evaluated in the UFSAR that are of potential interest are:

- Excess Steam Flow
- Loss of Load
- Loss of Offsite Power
- High Energy Line Break
- Flooding
- Turbine-Generator Trip

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

<u>Excess Steam Flow/Loss of Load</u>

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. A load rejection (or loss of load event) could occur if there were a decrease in steam flow.

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 4.1.2) 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 0.67% decrease in the normal full-power cold reheat flow from the high pressure (HP) turbine to the moisture separator reheaters (MSRs) and an approximately 0.76% 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 slight reduction (approximately 0.43%) in the MWe 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 a loss of load event.

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.

An electrical transient analysis (see Reference 3, Section 4.1.3) has determined that sudden changes in the electric power supplied to the HTEF would cause the balance between the electric power provided by the main generator to the HTEF and to the switchyard and transmission system to be readjusted, with the main generator and switchyard remaining stable. These changes would be within the normal operational capabilities of the turbine control system and the other plant control systems 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 either an excess steam flow event or a loss of load event.

<u>Loss of Offsite Power</u>

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. Since 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.

High Energy Line Break

The routing of steam in the new piping (10" maximum diameter) from the cold reheat piping to the new reboiler 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 plants are designed to ASME codes and standards for piping, such that failures in steam piping and components at a nuclear plant 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. Since 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.

Flooding

The new piping and components from the new reboiler to the condenser and the new demineralized water tank 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-Generator Trip

The installation of new electrical devices at the high-voltage side of the main transformer 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. Existing high-voltage electrical equipment is designed to the codes and standards appropriate for this application, such that these types of faults or failures are exceedingly rare. Since the new electrical equipment will likewise be designed to the appropriate codes and standards, 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 turbine-generator trip.

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 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 introduce 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 will not 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. Select information from the new equipment and controls will also be 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 which 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.

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.

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.

The UFSAR for the reference plant used in developing the present 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 (July 1981), 2 (April 2013), and 3 (November 2021, 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 has not been 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.

Operator errors involving the new operator controls for steam and electric power to the HTEF could initiate, at most, slight transients in balance-of-plant systems. As discussed in the response to Question 1, 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.

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 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 minor changes in the flow of steam, condensate, demineralized water, or electrical power. 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. Since 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. REFERENCES

1. INL/EXT-20-60104, Rev. 0, *"Probabilistic Risk Assessment of a Light Water Reactor Coupled with a High-Temperature Electrolysis Hydrogen Production Plant,"* Vedros/Christian/Rabiti, October 2020.

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, April 2013.

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

Appendix 2
Design Case Assumptions

Reference NPP Assumption

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

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 H2 Generation Plant Design will be at the low (medium voltage) side of the step-down transformer to the H2 Generation Facility
- Electrical Distribution design required for the H2 Generation Plant beyond the demarcation is assumed inclusive of the H2 Generation Plant Design
- Modeling tools will confirm acceptable NPP electrical integration basis and voltage drop acceptability of power take-off to the demarcation.

Thermohydraulic

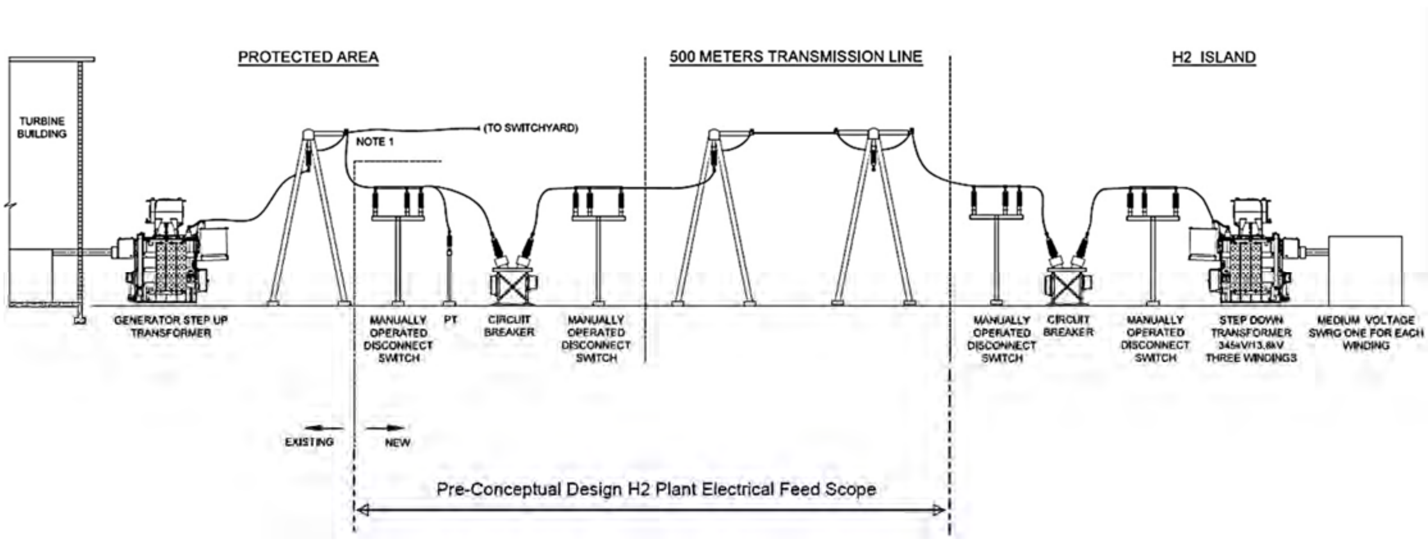
- Steam Input Requirement to the Hydrogen Island
  - 300°F steam at 50 psig at the H_2 Plant Demarcation Border
  - The value chosen is assumed to be bounding for the different H_2 system technologies.
- 25 MW thermal power extraction from the steam cycle
  - This value accounts for the projected need 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.

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.

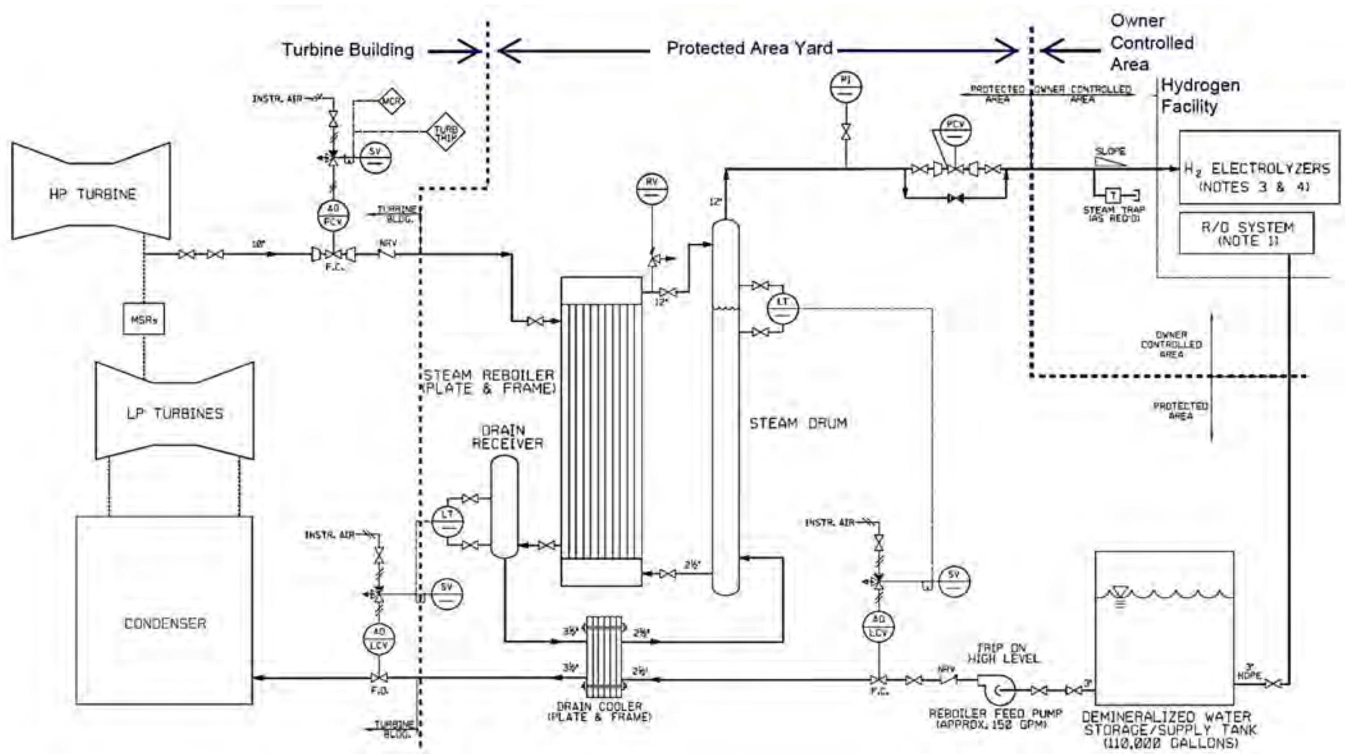
Electrical Design

- 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.



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.



Control Systems Design

To facilitate HTEF operation, a dedicated set of operator controls with remote HMI will be provided. The HMI will allow for control, indication, and alarm of the H_2 power line and steam supply; these controls will be electrically and functionally isolated from NPP controls, but the remote HMI will be collocated in the NPP Main Control Room. Existing plant fiber optic infrastructure will be used to communicate between the HMI and equipment associated with H_2 power line and steam supply. This permits the status of the HTE process parameters to be available to NPP Control Room operators to evaluate the impact of HTEF loading on NPP operation. It also allows necessary on/off control for operators to enable or isolate the HTEF supply steam and electrical power. Additional indication and controls will be provided local to the HSS equipment.

The operator should be trained in operating the power and steam supplies from the NPP to the H_2 plant using the new standalone HMI. A special procedure should be prepared for this operation.

The following process parameters are expected to be available to allow plant personnel to monitor performance of the thermal and electrical extraction systems:

- Electrical power consumption on the plant computer logging system
- Steam flow diverted from the plant on the plant computer system (for plant performance engineer)
- HSS equipment trouble alarm in Main Control Room
- Hydrogen plant trip or fire alarm in Main Control Room

Appendix 3
Advisory Subcommittees Excerpts

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 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 [2] 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
    - 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 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 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 and RG 1.174.

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 nuclear power plant's (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 USNRC 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 that:

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, and
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 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 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 Failure Modes and Effects Analysis (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:

- 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 and
- 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 in shown in Table 1. The most susceptible component is the transmission tower in the switchyard with a 0.8 probability of failure at 0.2 psi.

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 a PWR and the other 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 two-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.

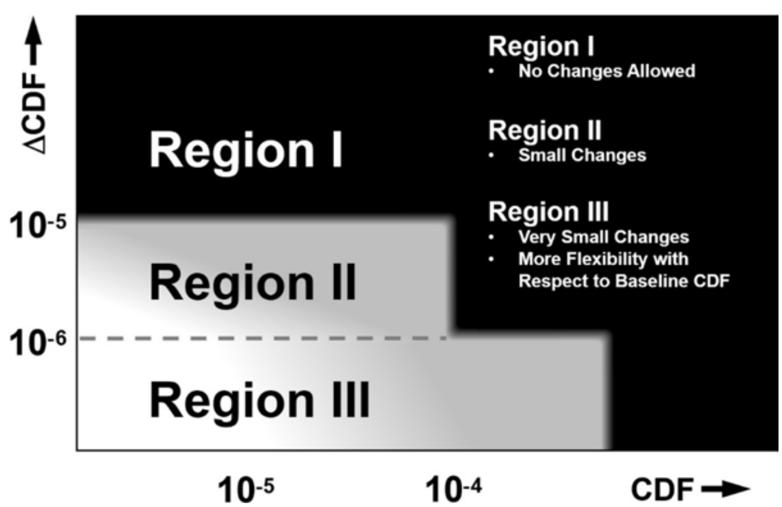
For the bounding large HTEF, PWR PRA results showed the initial IE frequency for main steam line break (MSLB) went up by 5.6% from the initial value. The switchyard related LOOP initiating event frequency increased by 1%. The bounding large HTEF with PWR CDF increased a minimal 6.56% from 8.33E-06 to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

As described in RG 1.174 and shown in Figure 11, 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, the 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.



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 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 in Figure 11.

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 shown in Figure 11.

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 toa prominent industry Computer Aided Fault Tree Analysis system (CAFTA) model
- It is important to modify the model to match the current design of the heat extraction system (HES) and the HTEF as details become available
- Consider hydrogen detonation seismic induced failures
- Model the effects of site layout, 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

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 (INPO) IER 17-5**Error! Reference source not found.**, 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 HTE 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 MWth 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 that 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 IES 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 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}, 300º 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 being operated 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 3rd 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 to be included in operator training programs.

4. Operations Environment:
   - Operation of the 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 that provides for and limits 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 based on plant design, 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 operators 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 to 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 to 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 off primary and secondary power mismatch.
   - The system should be started up / shut down via automation if possible
   - The system should have automatic isolation feature based off 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.
   - Pre-conceptual 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 RP 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 in that light.
   - North American Electric Reliability Corporation (NERC) requirements will play a role in how the electrical tie in is designed and 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 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 that 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 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.

Electrical and Switchyard Subcommittee

The purpose of the Electrical and Switchyard Subcommittee work was to ensure any new electrical power aspects 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/50.59:

- The HTE equipment's requirement for sufficient electrical supply from power plants will require modifications to the nuclear power plant's electrical infrastructure.
  - Those modifications require modification type 50.59 screening ensuring proper implementation reviews and documents are completed.
  - Many installation aspects of the electrical portions of the proposed equipment will categorize 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 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 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, grid fault contributions, etc. each are expected to change and could be subject to 50.59 review.
- Many nuclear plant facilities take advantage on main generator power during the initial moments of a 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 main generator trip.
  - As the HTE project interconnects electrical connections & components in these areas, the licensee shall review adverse potential aspects of adding these connections and components under the 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 includes but is not limited to: description of new components & 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 50.59 review if the nuclear operations team will be required to operate and monitor the new components and connections. The licensee will likely be required to review under 50.59 the added activities to the operators including:
  - 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 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

Control Systems Subcommittee

The purpose of the Control Systems Subcommittee is to ensure that all new control aspects needed to ensure 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:
  - 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 main control room (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 demin water to operate.
- To keep controls as simple as possible, only a few items are needed for the main control room 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 CRF 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/cyber process rules are not expected to challenge the results of the 50.59.

Regulatory Strategy Subcommittee

The Regulatory Strategy Subcommittee informed the AE design activities related to USNRC 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 an LAR for prior USNRC 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 and inform the AE of additional USNRC regulatory requirements that could, based on unique plant specific CLBs and drive the need for additional USNRC 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				Aligns with 50.59 criterion 5
	Involve significant reduction in margin of safety				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

Rule	Criteria	Steam Diversion	Electrical Diversion	Facility Location	Comments
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: Technical specification 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 citing of the HTE equipment. A TS revision relative to the electrical design should not require a TS change provided power remains assured 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 location of the HTE equipment 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.

4. 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 USNRC 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 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 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:
  - 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.