Guidelines for

10 CFR 50.59 Screenings and Evaluations For Non-power Reactors

PURPOSE

This document is based on the Nuclear Energy Institute (NEI) 96-07 R1, *Guidelines for 10 CFR 50.59 Implementation* ~~approved~~ endorsed by the NRC in Regulatory Guide (RG) 1.187 Revision 2 and Electric Power Research Institute (EPRI)/Nuclear Energy Institute (NEI) Joint Task Force report, "Guideline on Licensing Digital Upgrades: EPRI TR-102348, Revision 1, NEI 01-01, endorsed by Regulatory Issue Summary (RIS) 2002-22 and RIC 2002-22 Supplement 1. There are fundamental differences in regulatory requirements for research and test reactors (RTRs) as compared to nuclear power reactors; this guidance is intended to focus ~~research and test reactor~~RTR licensees on those parts of the NEI guidance that are most relevant. ~~The~~ This document is intended to provide terminology and examples specific to RTR licensees ~~of~~ based on the NEI document, which is intended to support the nuclear power plant environment.~~, and d~~Definitions used ~~in~~ from the NEI document with which RTR licensees might not be familiar are included in the definitions section below.

10 CFR 50.59 provides a method for evaluating changes to the facility and tests or experiments to determine if the activities can be accomplished without prior NRC approval. If an activity requires prior NRC approval, the activity must be approved by the NRC via license amendment in accordance with 10 CFR prior to implementation. Thus, 10 CFR 50.59 provides a threshold for regulatory review -- not the final determination of safety -- for proposed activities.

Determining that a proposed activity requires prior NRC approval does not determine whether it is safe. In fact, a proposed activity that requires prior NRC approval may significantly enhance overall plant safety at the expense of a small adverse impact in a specific area. It is the responsibility of the ~~RTR~~ licensee to assure that proposed activities are safe, and it is the role of the NRC to confirm the safety of those activities that are determined to require prior NRC review.

As reflected in NEI 01-01, endorsed by RIS 2002-22, the 10 CFR 50.59 process applies to digital upgrades as it does to other plant modifications. However, there are specific considerations that should be addressed including, for example, different potential failure modes of digital equipment as opposed to the equipment being replaced, the effect of combining functions of previously separate devices into one digital device, and the potential for software common cause failures. These digital considerations are addressed in the design process, including in failure analyses and other engineering evaluations.

1. **Definitions** Note: all definitions are from NEI-96-07 Revision 1 unless otherwise noted.

1.1. 10 CFR 50.59 Changes, Tests and Experiments

Definition:

10 CFR 50.59 Changes, Tests and Experiments is the regulation that establishes the conditions under which licensees may make changes to the facility or procedures and conduct tests or experiments without prior NRC approval, i.e. without an amendment to the facility operating license of the Technical Specifications. 10 CFR 50.59 focuses on the effects of proposed activities on the safety analyses that are contained in the updated FSAR and are a cornerstone of each Research and Test Reactor's (RTR's) licensing basis. In addition to 10 CFR 50.59 control of changes affecting the safety analyses, there are several other complementary processes for controlling activities that affect other aspects of the licensing basis, including:

- Amendments to the operating license (including the Technical Specifications) are sought ~~and obtained~~ under 10 CFR 50.90.
- Where changes to facility or procedures are controlled by more specific regulations (e.g. ~~quality assurance,~~ security and emergency preparedness program changes controlled under 10 CFR 50.54 ~~(a),~~ (p) and (q), respectively), 10 CFR 50.59 states that the more specific regulation applies.
- Changes that require an exemption from a regulation are processed in accordance with 10 CFR 50.12.
- Guidance for controlling changes to licensee commitments is provided by NEI 99-04, *Guideline for Managing NRC Commitment Changes*.
- Where a licensee possesses a license condition that specifically permits changes to the NRC-approved fire protection program (i.e., has received the standard fire protection license condition contained in Generic Letter 86-10), subsequent changes to the fire protection program would be controlled under the license condition and not 10 CFR 50.59.

1.2. 10 CFR 50.59 Evaluation

Definition:

A 10 CFR 50.59 evaluation is the RTR's documented evaluation against the 10 CFR 50.59(c)(2) criteria. Results of the 10 CFR 50.59 evaluation will determine if the proposed change, test, or experiment requires NRC review and approval prior to implementation under 10 CFR 50.90. A 10 CFR 50.59 evaluation is normally performed after the RTR's screening process "screens into" the evaluation. However, an RTR can start with the 10 CFR 50.59 evaluation if it is obvious that a screening would "screen into" an evaluation.

1.3. 10 CFR 50.59 Screening

Definition:

A 10 CFR 50.59 screening is a process of determining if a 10 CFR 50.59 evaluation shall be performed. The 10 CFR 50.59 screen determines if the change, test, or experiment has the potential to have an adverse impact to the facility, as described in the Final Safety Analysis Report (FSAR):

- To a design function of a system, structure, or component (SSC)
- To a method of performing a design function

Commented [HD1]: In order to endorse TRTR documents, the document will have to be more precise on its language and references to prevent uncertainty in requirements vs. guidance. A list of References with the applicable versions is recommended for inclusion.

Commented [HD2R1]: The version/revision of other documents referenced should be included since the subject references *could* subsequently change to be contrary to the original NRC endorsement.

Commented [BP3]: I would suggest rewording to reflect real differences between the regulatory requirements associated with NPRs versus NPPs.

Commented [HD4R3]: The 50.59 guidance for NPPs is fragmented between many documents. It is preferable to have one RTR document that compiles and harmonizes the various guidance documents into a single-useable document for RTRs.

Commented [HD5]: - Responsibility should be assigned to licensee. The physical facility is not the licensee.

Commented [NC6]: If copying definitions from 96-07, copy the entire definition. Partial copies could distort the meaning and result in uncertainty.

Commented [HD7R6]: Additional definitions from 96.07 should be considered for inclusion to facilitate discussions within the document (e.g., screen and adverse)

Commented [BP8]: Appendix B quality assurance does NOT apply to RTRs. QAPs are included in the TSs consistent with ANS 15.1 and NUREG-1537

Commented [BP9]: Deleted - 10 CFR 50.54(a)(1) applies to nuclear power plant and fuel reprocessing plant licensees subject to the quality assurance criteria in appendix B

Commented [BP10]: No currently licensed RTRs have an approved fire protection plan –

Commented [HD11R10]: If, in a general sense this provision is included by TRTR to be future inclusive, the NRC staff have no objection.

Commented [SW12R11]: SHINE and some other applicants have committed to addressing some GDCs in their application, including fire protection programs.

- To an evaluation for demonstrating that intended design functions will be accomplished
- To a design basis limit for a fission product barrier
- ~~A Technical Specification change is required. [*note: technical specification bases are not license requirements, they are support for other license documentation. References in bases to analyses in the FSAR should also undergo a 50.59 evaluation*]~~

Commented [BP13]: Deleted – does not fit outline (the bullets above all start with "To a"). Additionally, If a TS change is required, including a change to the TS bases a TS amendment is required and 50.59 cannot be utilized.

Commented [HD14R13]: Structure of [ANS/ANSI/15.1] TS bases require amendment because the bases are included in the TS and form a part of the license. Suggestion to address this concept is in the regulatory basis subsection of the Applicability section below.

~~1.5.~~1.4. Accident

Definition:

An accident is a transient that causes a design basis accident previously evaluated in the Updated FSAR (UFSAR), exceeding the facility's design basis limit.

"Accidents of a different type" are either:

1. Credible accidents that the proposed activity could create ~~which are not bounded by UFSAR-evaluated accidents~~. ~~The postulated accident may be similar to an analyzed accident, but the frequency of the initiating event or the consequences may be sufficiently different that the accident is considered a different type than considered by the UFSAR analysis.~~ [NEI 96-07 R1], or

2. Accidents that the proposed activity could create which were not previously identified as credible [REG GUIDE 1.187]

Commented [CN15]: The newest version of RG1.187 make it clear that "bounding" is not the appropriate criteria.

Commented [CN16]: Frequency and consequences are addressed by other questions explicitly. The idea is that an accident of a different type is fundamentally different in type not frequency or consequences.

"Accidents previously evaluated in the ~~U~~FSAR" are anticipated (or 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. For purposes of 10 CFR 50.59, the term "accidents" encompasses other events for which the plant is required to cope and that are described in the ~~U~~FSAR (e.g., fire, earthquakes and flooding). Accidents also include new transients or postulated events added to the licensing basis based on new NRC requirements. [NEI 96-07 R1]

Commented [HD17]: The NRC staff suggest that the terminology related to FSARs (e.g., updated FSAR, FSAR updates , and the FSAR (as updated)) be uniformly applied and conform to the regulations [50.59] and to the terminology used in RG 2.7 of the NPUF rule (e.g., 50.719(e)).

Commented [BP18]: Need to ensure consistent and applicable terminology; For clarity, the document should include related definitions for MHA and reflect related NUREG-1537 terminology.

A **Design-basis accident** (or event) is **postulated accident** that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety.

Similarly; a **Beyond-design-basis events** can reduce or eliminate the margin of safety of the structures, systems and components, possibly resulting in a catastrophic failure (e.g., an accident more severe than the plant was designed to accommodate

~~1.6.~~1.5. Change

Definition:

Change means a modification or addition to, or removal from, the facility or procedures that affects (1) a design function, (2) method of performing or controlling the function, or (3) an evaluation that demonstrates that intended functions will be accomplished. Replacing equipment with an exact like for like replacement is not considered a change, thus 10 CFR 50.59 does not apply.

Commented [BP19]: Suggest a global change is to remove reference to "the plant" and replace with facility and/or reactor as appropriate.

Commented [BP20]: The NRC staff recommend including a specific "*exact like-for-like*" definition to clarify. A definition should be provided to address misinterpretation. For example, replacing a paper chart recorder that has pens with a paperless digital recorder is not like for like, even though it is a new chart recorder replacing an old one.

~~1.7.~~1.6. Common Cause Failure

Definition:

Failures of equipment or systems that occur as a consequence of the same cause. The term is usually used with reference to redundant equipment or systems or to uses of identical equipment in multiple systems. Common cause failures can occur due to design, operational, environmental, or human factor initiators. Common cause failures in redundant systems compromise safety if the failures are *concurrent failures*, that is, failures which occur over a time interval during which it is not plausible that the failures would be corrected.

~~1.8.~~1.7. Departure from a method of evaluation described in the ~~U~~FSAR (as updated)

Definition:

Departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses means:

- Changing any of the elements of the method described in the FSAR (as updated) unless the results of the analysis are conservative or essentially the same; or
- Changing from a method described in the FSAR to another method unless that method has been approved by NRC for the intended application.

Commented [BP21]: Be consistent throughout; The definition does not match the 1.7 title (FSAR as updated vs. UFSAR)

~~1.9.~~1.8. Design Bases

Definition:

Design bases means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.
The basis for the facility's Safety Limit (SL), Limited Safety System Settings (LSSS), and Limiting Conditions of Operations are Design Bases.

"Design functions" are UFSAR-described design bases functions and other SSC functions described in the UFSAR that support or impact design bases functions. Implicitly included within the meaning of design function are the conditions under which intended functions are required to be performed, such as equipment response times, process conditions, equipment qualification and single failure. [NEI 96-07 R1, NEI-01-01]

"Design bases functions" are functions performed by systems, structures and components (SSCs) that are (1) required by, or otherwise necessary to comply with, regulations, license conditions, orders or technical specifications, or (2) credited in licensee safety analyses to meet NRC requirements. [NEI 96-07 R1]

~~1.10.~~1.9. Facility

Definition:

Facility as described in the final safety analysis report (as updated) means:

- The structures, systems, and components (SSC) that are described in the final safety analysis report (FSAR) (as updated),
- The design and performance requirements for such SSCs described in the FSAR (as updated), and
- The evaluations or methods of evaluation included in the FSAR (as updated) for such SSCs which demonstrate that their intended function(s) will be accomplished.

~~1.11.~~1.10. Final Safety Analysis Report (as updated)

Definition:

Final Safety Analysis Report (as updated) means the Final Safety Analysis Report submitted in accordance with Sec. 50.34, as amended and supplemented, and as updated per the requirements of Sec. 50.71(e).

Commented [CG22]: NRC staff recommends adopting the provisions of the NPUF Rule that will require FSAR updates.

~~1.12.~~1.11. Human- Systems Interface (HSI)

Definition:

All interfaces between the digital system and plant personnel including operators, maintenance technicians, and engineering personnel (e.g., display or control interfaces, test panels, configuration terminals, etc.). These interfaces include information and control resources used by plant personnel to perform their duties and tasks. Currently, HSI is the term that is synonymous with and replaces human machine interface (HMI) and man-machine interface (MMI). Principal HSIs are: alarms, information displays and controls. A HSI may be made up of hardware and software components and is characterized in terms of its physical and functional characteristics.

"**Methods of evaluation**" means the calculational framework used for evaluating behavior or response of the facility or an SSC. [NEI 96-07 R1]

Commented [BP23]: NRC staff recommend providing additional detail regarding the type of facility responses that are being considered. Typically, RTRs do not take credit for mitigation responses.

"**Method of performing or controlling a function"** means how a design function is accomplished as credited in the safety analyses, including specific operator actions, procedural step or sequence, or whether a specific function is to be initiated by manual versus automatic means. [NEI-96-07, 3.3]

"Implemented" an activity is considered implemented when it provides its intended function, that is, when it is placed in service and declared operable. Thus, a licensee may design, plan, install and test a modification prior to receiving the license amendment to the extent that these preliminary activities do not themselves require prior NRC approval under 10 CFR 50.59. [NEI 96-07 R1]

"Maintenance activities" include troubleshooting, calibration, refurbishment, maintenance-related testing, identical replacements, housekeeping, and may include temporary alterations to the facility or procedures that directly relate to and are necessary to support the maintenance. [NEI 96-07 R1 4.1.2]

"Malfunctions of SSCs" are generally postulated as potential single failures to evaluate ~~plant~~ facility performance with the focus being on the result of the malfunction rather than the cause or type of malfunction. [NEI 96-07 R1 4.3.6]

~~1.13.~~1.12. Procedures

Definition:

Procedures as described in the Final Safety Analysis Report (as updated) means those procedures that contain information described in the FSAR (as updated) such as how structures, systems, and components are operated and controlled (including assumed operator actions and response times).

1.13 Safety Analyses

Definition:

Safety analyses are the analyses performed pursuant to license application, or renewal, to demonstrate the integrity of the fission product boundaries, the capability to shut down the reactor and maintain the reactor in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures. Safety analyses are presented in the ~~U~~FSAR.

1.14 **"Tests or experiments"** not described in the ~~final safety analysis report~~FSAR (as updated) means any activity where any structure, system, or component is utilized or controlled in a manner which is either:

(i) Outside the reference bounds of the design bases as described in the ~~final safety analysis report~~FSAR (as updated) or

(ii) Inconsistent with the analyses or descriptions in the ~~final safety analysis report~~FSAR (as updated). [10 CFR 50.59]

APPLICABILITY

Licensees should first determine applicability of 10 CFR 50.59 for the change(s) being made and then perform screen activities to determine if 10 CFR 50.59 evaluations are required.

Regulatory Basis

As stated in Section (b) of 10 CFR 50.59, the rule applies to each holder of a license authorizing operation of a production or utilization facility, including a non-power production or utilization facility that has permanently ceased operations.

Per 10 CFR 50.59(c)(1)(i), proposed activities that require a change to the technical specifications (TS) must be made via the license amendment process, 10 CFR 50.90. TS Bases are not part of TS per 10 CFR 50.36(a)(1). However, since RTRs do not have a separate bases document, changes to the bases portions of the technical specifications cannot be made under 10 CFR 50.59. Aspects of proposed activities that are not directly related to the required technical specification change are subject to 10 CFR 50.59.

To reduce duplication of effort, 10 CFR 50.59(c)(4) specifically excludes from the scope of 10 CFR 50.59 changes to the facility or procedures that are controlled by other more specific requirements and criteria established by regulation. For example, 10 CFR 50.54, which was promulgated after 10 CFR 50.59, specifies criteria and reporting requirements for changing physical security and emergency plans.

Process Guidance

The 10 CFR 50.59 process applies to

- tests or experiments not described in the ~~U~~FSAR,
- changes to the facility or procedures as described in the ~~U~~FSAR,
- changes made in response to new requirements or generic communications, and
- temporary changes proposed as compensatory actions for degraded or nonconforming conditions.

If a proposed activity requires multiple elements (such as other modifications, temporary modifications, procedures required to support the change), each element of the proposed activity and concurrent changes must be screened or evaluated (for adverse effects) unless

(1) They are interdependent as in the case where a modification to a system or component necessitates additional changes to other systems or procedures; or

(2) They are performed collectively to address a design or operational issue.

If concurrent changes are being made that are not linked, each must be evaluated separately and independently of each other.

Changes to the facility or procedures that are controlled by other more specific requirements and criteria established by regulation are excluded from the scope of 10 CFR 50.59; aspects of proposed activities that are not directly related to the required change are subject to 10 CFR 50.59. However, if the change conducted under other regulations involves activities, systems or components not assessed or regulated by the applicable 10 CFR regulation, 50.59 would apply*; for example, the installation of new equipment to meet the requirements of 10CFR20 would be evaluated to meet performance criteria, but the installation of a new power source for this equipment would typically not be considered by 10CFR20 and therefore 10CFR50.59 applies*

- 10 CFR 50.90, License and Technical Specifications
- ~~10 CFR 50.54(a), Quality Assurance~~
- 10 CFR 50.54(p), physical security
- 10 CFR 50.54(q), emergency plans
- ~~10 CFR 50.55a (Codes and Standards)~~
- CFR 50.12, (Specific Exemptions)
- CFR Part 20 (Standards for Radiation Protection).
- To the extent the ~~U~~FSAR changes are directly related to the activity implemented via another regulation, applying 10 CFR 50.59 is not required.

~~U~~FSAR Modifications that are not the result of activities performed under 10 CFR 50.59 are not subject to control under 10 CFR 50.59. Such modifications include:

Commented [HD24]: The NRC staff recommend that additional wording from 96-07 be incorporated to ensure clarity of applicability and the regulatory basis for stated rationale.

Commented [HD25R24]: As an example, we have incorporated 96-07 words from section 4.1 and 4.1.1

Commented [HD26]: This language is based on the NPUF rule

Commented [BP27]: 50.54 (a) is applicable to NPPs and fuel reprocessing facilities only. (not applicable to RTRs)

Commented [HD28]: Not applicable to RTRs

- Reformatting and simplification of ~~U~~FSAR information
- Removal of obsolete or redundant information and excessive detail
- Editorial changes to the ~~U~~FSAR (including referenced procedures, topical reports, etc.)
- Clarifications to improve reader understanding
- Correction of inconsistencies within the ~~U~~FSAR (e.g., between sections)
- Minor corrections to drawings, e.g., correcting mislabeled valves
- Similar changes to ~~U~~FSAR information that do not change the meaning or substance of information presented.

Maintenance: ~~Research reactor~~RTRs are not regulated under 10CFR50.65 and maintenance activities and procedures are therefore generally subject to 10CFR50.59, however:

- Maintenance activities that do not permanently alter the design, performance requirements, operation or control of SSCs are not subject to 10CFR50.59 as long as ~~plant~~ facility conditions are such that the design function is not required
- 10CFR50.59 does not apply to maintenance actions which restore the equipment and system to the as-designed condition of the ~~U~~FSAR, including:
  - post-maintenance testing
  - preventative maintenance and testing activities conducted using approved procedures
  - Corrective maintenance that replaces failed components with equivalent functional components so as to not alter the design, performance requirements, operation or control of the SSC
    - Replacement with a component that does not have equivalent characteristics, failure modes, and reliability as the original is a design change, and requires application of 10CFR50.59

NOTE: Commercial off the shelf components that do not have documented failure modes and reliability assessment determined to be equivalent functional components based on a minimum of:

- Comparing manufacturer or vendor information for the original and the replacement component,
- Comparing established or standard use for the replacement component with service conditions of original component, and
- Testing (to the extent feasible).

CAUTION: Instrumentation and components in general have operating characteristics valid over specific ranges; using components or instruments with larger or smaller ranges than the original may degrade performance by positioning normal readings or protective setpoints outside the recommended range of use.

- If corrective maintenance includes interim compensatory action to address the condition and involves a temporary procedure or facility change, 10 CFR 50.59 should be applied to the temporary change.

NOTE: The intent is to determine whether the temporary change/compensatory action impacts other aspects of the facility or procedures described in the UFSAR, not to evaluate the degraded condition

2. 10 CFR 50.59 Screening Process

Once it has been determined that 10 CFR 50.59 is applicable to a proposed activity, screening is performed to determine if the activity should be evaluated against the evaluation criteria of 10 CFR 50.59(c)(2).

Sections xxx and xxx below provide guidance and examples for determining whether an activity is (1) a change to the facility or procedures as described in the UFSAR or (2) a test or experiment not described in the UFSAR (i.e., it screens in), and should be evaluated against the evaluation criteria of 10 CFR 50.59(c)(2). If an activity is determined to be neither, then it screens out and may be implemented without further evaluation under 10 CFR 50.59. Activities that are screened out from further evaluation under 10 CFR 50.59 should be documented as discussed in Section 4.2.3.

Activities that screen out may nonetheless require FSAR information to be updated. Licensees should provide updated FSAR information to the NRC in accordance with 10 CFR 50.71(e).

2.1. Objective:

2.1.1. The use of the screening process is to determine if a ~~full~~ 10 CFR 50.59c (2) evaluation ~~should occur~~is required. This is alleviating the administrative burden on the staff for changes that will not have an adverse impact on the facility as described in the FSAR. The screening of a change to the facility allows the evaluator to quickly assess the project for adverse effects without having to engage in a great deal of detail. If there is a potential for an adverse effect on the description in the FSAR, then the evaluation moves into the analysis against the c(2) criteria of the 10 CFR 50.59.

2.2. Process

Commented [OP29]: This "new" guidance for how RTRs might determine if a COTS component is "like-for-like" is not present (as such) in 96-07. The NRC staff considers the presented information necessary, but not sufficient for determining acceptable use. The regulations in 10 CFR Part 21 should be considered in defining a basic component

Commented [HD30R29]: Like for like components are not considered a changed under 50.59; however, this guidance is not, in itself, sufficient for an adequate like for like determination.

Commented [HD31R29]: Additionally, Section 5.0 of NEI 96-07) on engineering evaluation is suggested for adaptation to TRTR guidance:

Commented [HD32R29]: 96-07 does not include technical guidance. I f it is to be included in this document it needs to be developed further.

Commented [NC33]: NRC staff suggests adding the pertinent text From NEI 96-07 R1.

Commented [HD34]: This would be Section 4.21 and 4.2.2 in 96-07, R1

Commented [NC35]: The NRC staff recommend that TRTR insert a paragraph that states the intent of screens vs. 50.59 evaluations. Screening is about whether or not the proposed change is adverse and therefore must be further evaluated. It is not the intent of screening to try to evaluate the magnitude of the adversity.
In contrast the 50.59 evaluation is about magnitude of the effect and a determination if the change is permissible without prior NRC approval.

2.2.1. Each RTR should have a 50.59 screening form with the following Screening Criteria (*see below*): The answers to the questions in this section should be fully supported by the information provided in the Description Summary. A yes to any of the questions below will result in an analysis based on the 10 CFR 50.59 criteria.

2.2.1.1. Does the change ~~have the potential to~~ adversely affect a design function of a System, Structure, or Component (SSC) as described in the FSAR?

Commented [HD36]: Generally, the rule change to 50.59 that took effect in 1999 eliminated "potential". The reason is that more certainty is required.

- Does the activity decrease the reliability of an SSC design function, including either functions whose failure would initiate a transient/ accident or functions that are relied upon for mitigation?
- Does the activity reduce existing independence, redundancy, diversity, or defense-in-depth?
- Does the activity add or delete an automatic or manual design function of the SSC?
- Does the activity convert a feature that was automatic to manual or vice versa?
- Does the activity introduce an unwanted or previously unreviewed system or materials interaction?
- Does the activity adversely affect the ability or response time to perform required actions, e.g., alter equipment access or add steps necessary for performing tasks?
- Does the activity degrade the seismic or environmental qualification of the SSC?

Example: Replacement of a component such as a shim arm drive mechanism that has a slower release of a clutch than its predecessor. Using a similar clutch may not end up being a true one-for-one replacement. As this has the potential to affect a design function, it would screen in *and would require a full analysis against the criteria of 10 CFR 50.59. If the criteria are not met, then a license amendment would need to be filed with the NRC.*

2.2.1.2. Does the change ~~have the potential to~~ adversely affect the method of performing the intended design function?

Example: Replacing a mechanically operated actuator with an air operated actuator for a ventilation damper utilized for building closure. As the potential exists to adversely affect the method of performing the intended design function, this would screen in *and require a full 50.59 evaluation.*

2.2.1.3. Does the change ~~have the potential to~~ adversely affect an evaluation (i.e. computational method) that demonstrates the intended design function will be accomplished?

Commented [HD37]: "Potential" is problematic phrasing

Example: Changing the model used for an accident analysis. If the changes in modeling are within the constraints and limitations associated with the use of the method (e.g. identified in a topical report and/or SER), the changes would screen out *of a 50.59 evaluation. Otherwise, an evaluation under 10CFR 50.59(c)(2)(viii) [see section 4.7 below] would be required. Changes only to methods of evaluation do not require evaluation against the other seven 50.59 criteria.*

2.2.1.4. Is this a change (either positive or negative) to a design basis limit for a fission product barrier (i.e. cladding, primary coolant boundary, or confinement)?

Example: Creating a new penetration in the confinement or containment wall. If the potential exists for changing a design basis limit, any changes to a fission product barrier require a 50.59 evaluation under 10CFR 50.59(c)(2)(vii) [see section 4.6 below].

2.2.1.5. Does the proposed change have any adverse effect on the design function of the SSCs or accident analyses described in the FSAR?

Commented [CG38]: Recommend TRTR add Screening example here

2.2.1.6. Do any sub-activities under the overall proposed change result in adverse effects as described above? (i.e. Does the change have an indirect effect on electrical distribution, structural integrity, environmental conditions or other UFSAR-described design functions?) If yes, such activities should be evaluated separately if embedded in a larger engineering change.

2.2.1.7. Is the activity a test or experiment not described in the FSAR?

- Is an SSC utilized or controlled in a manner that is outside the reference bounds of the design for that SSC or inconsistent with analyses or description in the ~~U~~FSAR?
- Is the test or experiment not bounded by tests and experiments that are described in the ~~U~~FSAR?
- If the test or experiment is not described in the ~~U~~FSAR, will affected SSCs need to be appropriately isolated from the facility?

3. 10 CFR 50.59 Criteria Evaluation Process

If the activity is a change to the Facility or Procedures as described in the ~~U~~FSAR and ~~-~~if the proposed change does not screen out, then 10 CFR 50.59 evaluation is required to determine whether such changes adversely affect a UFSAR-described design function.

The 10 CFR 50.59 evaluation determines whether the effects of proposed activities on accidents and malfunctions previously evaluated in the UFSAR and their potential to cause accidents or malfunctions whose effects are not bounded by previous analyses. The written evaluation must address the applicable criteria of 10 CFR 50.59(c)(2), Each activity must be evaluated against each applicable criterion.

The effects of a proposed activity being evaluated under 10 CFR 50.59 should be assessed against each of the evaluation criteria separately. For example, an increase in frequency/likelihood of occurrence cannot be compensated for by additional mitigation of consequences. Evaluations should consider the effects of the proposed activity on operator actions.

If any of the criteria are met, the licensee must apply for and obtain a license amendment per 10 CFR 50.90 before implementing the activity.

3.1. Each RTR should have a 50.59 evaluation form that assesses the change against the eight (8) criteria of 10 CFR 50.59(c)(2) and any justification of any Technical Specification changes.

3.2. Responsible Individuals shall consider if the change is a digital upgrade to an instrument and control system for an SSC.

3.2.1. Evaluations including digital upgrades should be evaluated against the Technical Specifications and the eight criteria of 10 CFR 50.59.

3.2.1.1. Any evaluation that requires a 'Yes' shall be submitted to the NRC as a license amendment.

3.2.2. Evaluations that include digital upgrades should consider the unique failure modes digital equipment will add to an SSC. Responsible individuals should evaluate new failure modes and understand the differences between the new and old equipment failure modes. Tables 1 and 2 of RIS 2002-22 Supplement 1 is a resource that should be used to make these evaluations. The following shall be considered in the evaluation:

3.2.2.1. Defense in depth of equipment within the boundaries of the equipment's intended purpose.

3.2.2.2. Common Cause Failures (i.e. identical software across protective equipment) shall be evaluated and mitigated.

3.2.2.3. Engineering evaluations shall address all the potential failure modes

3.2.2.4. Any evaluation that requires a 'Yes' to the eight criteria of 10 CFR 50.59 or any changes to Technical Specifications shall be submitted to the NRC as a license amendment.

4. Evaluation Guidance

Much of the guidance below is taken from NEI-96-07, "Guidelines for 10 CFR 50.59 Evaluations" Revision 1 and NEI-96-07 Appendix D, "Supplemental Guidance for Application of 10 CFR 50.59 to Digital Modifications", Revision 0, and summarized as applied to research and test reactors. The reader is referred to both documents for further details.

4.1. DOES THE ACTIVITY RESULT IN MORE THAN A MINIMAL INCREASE IN THE FREQUENCY OF OCCURRENCE OF AN ACCIDENT?

- ~~Is the effect of the proposed activity on the frequency of an accident discernable and/or attributable to the proposed activity?~~
- ~~Does the proposed activity change the frequency of occurrence to a more frequent category?~~

~~4.1.5.~~4.1.1. Reasonable engineering practices and engineering judgement should be used in determining whether the frequency of occurrence of an accident would more than minimally increase as a result of implementing a proposed activity. The effect of a proposed activity on the frequency of an accident must be discernable and attributable to the proposed activity in order to exceed the more than minimal increase standard. [ref: NEI-96-07 section 4.3.1]

~~4.1.6.~~4.1.2. For digital instrumentation, a common cause failure (CCF) evaluation should be made. If it is determined that there is no discernable increase in likelihood (see section 4.2.3 below) then ~~there~~ the activity does not result in more than a minimal ~~is also no~~ increase in frequency of occurrence of an accident.

> Example: Given the facility has evaluated for a loss of coolant flow accident, an upgrade to a main coolant pump and associated motor and motor controller to a variable frequency drive (VFD) control should be evaluated against this criterion. The number of failure modes associated with the VFD controller could increase the frequency of the loss of coolant flow accident. An evaluation of the failure modes and the associated likelihood should occur.
>
> Note: ~~, and t~~The facility should show how to mitigate the impacts of each failure mode, but this does not relate to this 50.59 question (it may affect other 50.59 evaluation questions).

> ~~Example: Given the facility has evaluated for a loss of coolant flow accident, an upgrade to a main coolant pump and associated motor and motor controller to a variable frequency drive (VFD) control should be evaluated against this criterion. The number of failure modes associated with the VFD controller could increase the frequency of the loss of coolant flow accident. An evaluation of the failure modes should occur, and the facility should show how to mitigate the impacts of each failure mode~~

Commented [NC39]: If these questions are to be retained, they need to be explained; i.e., explain what to do with the answer. Otherwise, delete them.

Commented [HD40R39]: For example, if the effect of the proposed activity on the frequency of an accident is discernable and/or attributable to the proposed activity, then the activity results in a more than minimal increase in the frequency of occurrence of an accident

Or

If the effect of the proposed activity on the frequency of an accident is NOT discernable and/or attributable to the proposed activity, then the activity DOES NOT result in a more than minimal increase in the frequency of occurrence of an accident.

Commented [NC41]: More guidance is needed. Seems to be saying that if you do not know, then it does not matter.

Commented [HD42R41]: Suggest that the text not use shorthand or incomplete reference to the 50.59 question

Commented [NC43]: Suggest revision to the example; the ability to mitigate impacts has nothing to do with the event likelihood.

Commented [NC44]: The NRC staff prefers that text boxes like this not be used. We cannot comment inside the box.
A similar look and feel can be achieved can be achieved by using borders to the paragraph in MS Word.

Commented [CG45]: NRC staff suggests replacing this example, as most research reactors do not have this system. More applicable system might be RPS, or CRDM related.

4.2. DOES THE ACTIVITY RESULT IN MORE THAN A MINIMAL INCREASE IN THE LIKELIHOOD OF OCCURRENCE OF A MALFUNCTION OF AN SSC IMPORTANT TO SAFETY?

- ~~Are the causes or failure modes of a malfunctions of the SSCs altered?~~
- ~~Has the likelihood of a malfunction of the important to safety SSCs increased more than minimally either directly or indirectly because of the proposed activity?~~

Commented [NC46]: If these questions are to be retained, they need to be explained; i.e., explain what to do with the answer. Otherwise, delete them. (See comment on Section 4.1)

4.2.1. The term 'malfunction of an SSC important to safety' refers to the failure of SSCs to perform their intended design functions – including both non-safety-related and safety-related SSCs. The cause and mode of a malfunction should be considered in determining whether there is a change in the likelihood of a malfunction.

4.2.2. After determining the effect of the proposed activity on the important to safety SSCs, a determination is made of whether the likelihood of a malfunction of the important to safety SSCs has increased more than minimally. Qualitative engineering judgement and/or an industry precedent is typically used to determine if there is more than a minimal increase in the likelihood of occurrence of a malfunction. A proposed activity is considered to have a negligible effect on the likelihood of a malfunction when a change in likelihood is so small or the uncertainties in determining whether a change in likelihood has occurred are such that it cannot be reasonably concluded that the likelihood has actually changed.

4.2.3. When considering digital instrumentation, the possibility of CCF must be evaluated. For the purposes of 50.59, a conclusion of *CCF Unlikely* is equivalent to a licensing condition of a CCF being not credible, and therefore not more than a minimal increase in likelihood. Alternately, a conclusion of *CCF Not Unlikely* is equivalent to a licensing condition of a CCF malfunction being credible, and the impact needs to be determined. However, an increase in the likelihood may or may not be "discernible" (see NEI 96-07, section 4.3.2)

Commented [CG47]: Should have reference to review of common cause failures, RIS 2002-22, Supplement 1 may provide some application guidance

Commented [NC48]: The terms "CCF Unlikely" and "CCF Not Unlikely" are from a very old draft of Appendix D. These terms are no longer used.

Commented [HD49R48]: Please update to "sufficiently low" in agreement with Appendix D to NEI 96-07, as endorsed by RG 1.187, Rev. 2.

Commented [CG50]: This reference is only for "discernable." Reference does not address CCF screening and results. This is misleading and should be revised.

Example: A facility upgrading the scram relays should evaluate against this criterion. The relays are part of the safety system and the failure modes should be evaluated to determine if there is more than a minimal increase in likelihood of occurrence of a malfunction if the relays had a faster failure rate. The facility should show how it would mitigate an increased failure rate. An example would be to replace the relays at half the expected life cycle.

4.3. DOES THE ACTIVITY RESULT IN MORE THAN A MINIMAL INCREASE IN THE CONSEQUENCES OF AN ACCIDENT?

- Will the proposed activity change, prevent or degrade the effectiveness of actions described or assumed in an accident discussed in the ~~U~~FSAR (as updated)?
- Will the proposed activity alter assumptions previously made in evaluating the radiological consequences of an accident described in the ~~U~~FSAR (as updated)?
- Will the proposed activity play a direct role in mitigating the radiological consequences of an accident describedin the ~~U~~FSAR (as updated)?
- Are the radiological consequences associated with the proposed activity exceed the bounds of the results of the UFSAR analysis?
- Are changes in radiological doses (to the public or personnel in the restricted areas) so small or the uncertainties in determining whether a change in consequences has occurred large enough that it can be reasonably concluded that the consequences have actually changed?

4.3.1. When determining which activities represent "more than a minimal increase in consequences," is must be recognized that the objective of the regulation is the protection of public health and safety. Therefore, an increase in consequences must involve an increase in radiological doses to the public or to ~~control room~~ facility staff~~operators~~. ~~This does not apply to occupational exposures resulting from routine operations, maintenance, testing, etc.~~ Where a change in dose consequences is so small or the uncertainties are such that it cannot be reasonably concluded that the consequences have actually changed, the change need not be considered an increase in consequences.

Commented [NC51]: Occupational exposure should be considered.

Example: For facilities that have ventilation systems in place to control release of effluents, a change to double the exhaust air flow should be evaluated against this criterion. A volumetric flow rate increase could increase the consequence of a radiological event prior to system realignment. The facility should evaluate the impact of the consequence and mitigate the consequence through engineer controls. This solution could be faster acting louvers or valves.

4.4. DOES THE ACTIVITY RESULT IN MORE THAN A MINIMAL INCREASE IN THE CONSEQUENCES OF A MALFUNCTION?

- Does the activity have potential to affect malfunctions evaluated in the ~~U~~FSAR (as updated) with radiological consequences?
- Does the activity increase radiological consequences and, if so, are they more than minimally increased?

4.4.1. Again, this involves an evaluation of radiological doses and the guidance for accidents in 4.3.1 applies to malfunctions.

Example: A facility that wants to replace a ventilation butterfly building isolation valve with a gate valve will have to evaluate against this criterion. The operation of the valve and the failure modes will have to evaluated to determine the consequences in an accident scenario. An example is the speed at which the valve would operate. If the valve closes at a slower rate the consequence of an equivalent accident would increase.

4.5. DOES THE ACTIVITY CREATE A POSSIBILITY FOR AN ACCIDENT OF A DIFFERENT TYPE?

- ~~Does the activity create possibility for a malfunction leading to an accident in the UFSAR?~~
- ~~Is the frequency of the malfunction that leads to an accident in the UFSAR higher than the frequency of the analyzed accident?~~
- ~~Does the activity create possibility for a malfunction leading to an accident that is not considered in the UFSAR?~~

Commented [HD52]: See previous comment in Section 4.1

4.5.1. From the main body of NEI 96-07, Section 4.3.5, the two considerations that need to be assessed when answering this question are *as likely to happen as* an *accident of a different type*. "~~The possible accidents of a different type are limited to those previously evaluated in the FSAR.~~ The accident must be credible in the sense of having been created within the range of assumptions previously considered in the licensing basis." If a qualitative assessment does not show the likelihood of occurrence is sufficiently low as to preclude consideration as above, then a determination must be made as to whether an accident of a different type exists (i.e. a new accident analysis is needed).

4.5.2. When considering digital instrumentation, the possibility of CCF must be evaluated. If it is concluded that a CCF is *not unlikely*, then there may be an accident of a different type. For the purposes of 50.59, a conclusion of *CCF Unlikely* is equivalent to a licensing condition of an accident initiator being NOT as likely to happen as those previously evaluated in the FSAR. Without a credible accident initiator, a new accident cannot be created due to a CCF. Alternately, a conclusion of *CCF Not Unlikely* is equivalent to a licensing condition of an accident initiator being as likely to happen as those previously evaluated in the FSAR (i.e. credible).

Commented [HD53]: Please update to "sufficiently low" in agreement with Appendix D to NEI 96-07, as endorsed by RG 1.187, Rev. 2.

Commented [NC54]: For CCFs, the criteria is much lower than. Refer to RIS 2002-22, Sup. 1.

4.5.3. HSI: If an adverse impact resulted from the physical interaction with the digital system/component, human error would be the potential accident initiator. If the human error can only cause accidents that have already been considered in the licensing basis, then an accident of a different type has not been created.

Example: A facility that is installing a new experimental facility that utilizes deuterium for cryogenic experiments should evaluate against this criterion. In this scenario the deuterium will be stored in an outside tank. Exposing the deuterium to a neutron flux will generate tritium creating the possibility of an accident that has not been previously evaluated. The facility will have to show that the limits of 10 CFR 20 will be met.

4.6 DOES THE ACTIVITY CREATE A POSSIBILITY FOR A MALFUNCTION OF AN SSC IMPORTANT TO SAFETY WITH A DIFFERENT RESULT?

- Does the malfunction involve an initiator or failure whose effects are not bounded by those explicitly described in the UFSAR?
- Does the malfunction have a different result or effect, i.e., is not bounded by that previously evaluated in the UFSAR?

Commented [HD55]: See previous comment in Section 4.1

4.6.1 A new failure mechanism is not a malfunction with a different result if the result or effect is the same as, or is bounded by, that previously evaluated in the FSAR.

4.6.2 CCF: Guidance is the same as discussed in section 4.2.3 above.

4.6.3 HSI: If an adverse impact resulted from the physical interaction with the digital system/component, human error would be the potential accident initiator and should be evaluated to determine if a different result could occur.

Example: A facility that wants to change out solenoid valves that operate valves important to safely aligning LOCA valve. If the solenoid valve had a failure mechanism that allowed it to fail in place, rather than full open or full shut, it could impact the operation of the LOCA valve. The facility will have to evaluate the failure mode against this criterion and mitigate the failure with an evaluation or engineered control.

4.7 DOES THE ACTIVITY RESULT IN A DESIGN BASIS LIMIT FOR A FISSION PRODUCT BARRIER BEING EXCEEDED OR ALTERED?

Commented [BP56]: Need to define design basis limit for RTRs. For example, fuel cladding temp. limit.

- Is the facility's predicted response with the activity implemented less conservative than the numerical design basis limit?
- Is the design basis limit changed with the activity implemented?

Commented [HD57]: See previous comment in Section 4.1

4.7.1 This evaluation focuses on the fission product barriers, if relevant and discussed in the FSAR: fuel cladding, reactor coolant system boundary, and containment/confinement. The change should be evaluated for any effect on the design basis limits set in the FSAR.

Example: If the facility wants to change the fuel cladding design this would constitute altering a fission product barrier. The facility would have to evaluate against this criterion and show that the change would mitigate the impact to the fission product barrier.

4.8 DOES THE 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?

- Are the results of analysis with the activity implemented yield nonconservative or not essentially the same as the results from the analyses of record?
- Is the new or different method of evaluation not approved by the NRC for the intended application?
- Is the method of evaluation not described, outlined or summarized in the UFSAR?
- Is the method of evaluation not a new NRC-approved methodology (e.g., new or upgraded computer code) to reduce uncertainty, provide more precise results or other reason, provided such use is (a) based on sound engineering practice, (b) appropriate for the intended application and (c) within the limitations of the applicable SER?

4.8.1 A determination is required before making changes to the evaluation methodologies described in the FSAR. In general, licensees can make changes to elements of a methodology without first obtaining a license amendment if the results are essentially the same as, or more conservative than, previous results. Similarly, licensees can also use different methods without first obtaining a license amendment if those methods have been approved by the NRC for the intended application.

Example: A facility that used MCNP software to evaluate their accident scenarios that wanted to use the latest version of the MCNP software should evaluate the software against the criterion. The facility should show that the software will evaluate the scenarios as well as the previous versions. This may be performing the evaluations on both platforms to determine the results from each are within an acceptable error.

DISPOSITION OF 10 CFR 50.59 EVALUATIONS

There are two possible conclusions to a 10 CFR 50.59 evaluation:

(1) The proposed activity may be implemented without prior NRC approval.

(2) The proposed activity requires prior NRC approval.

For proposed activities that are determined to require prior NRC approval, there are three possible options:

(1) Cancel the planned activity.

(2) Redesign the proposed activity so that the it may proceed without prior NRC approval.

(3) Apply for and obtain a license amendment under 10 CFR 50.90 prior to implementing the activity. Technical and licensing evaluations performed for such activities may be used as part of the basis for license amendment requests. [NEI 96-07, Section 4.5]

DOCUMENTATION AND REPORTING

10 CFR 50.59(d) requires the following documentation and recordkeeping:

The licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments made following screening out an evaluation, including a written evaluation that provides the bases for the determination that the change, test or experiment does not require a license amendment.

The records of changes in the facility must be maintained until the termination of a license issued pursuant to this part or the termination of a license issued pursuant to 10 CFR Part 54, whichever is later.

Records of changes in procedures and records of tests and experiments must be maintained for a period of 5 years.

A report must be submitted at intervals not to exceed 24 months containing a brief description of any changes, tests and experiments, and a summary evaluation against the eight criteria of 10 CFR 50.59(c)(2) that are determined not to require prior NRC approval.

Activities that were screened out, canceled or implemented via license amendment need not be included in this report.

Screening Documentation

Typically, the screening documentation is retained as part of the change package. This documentation does not constitute the record of changes required by 10 CFR 50.59, and thus is not subject to 10 CFR 50.59 documentation and reporting requirements, however it is recommended to keep such records for the life of the activity.

10 CFR 50.59 record-keeping requirements apply to 10 CFR 50.59 evaluations performed for activities that screened in, not to screening records for activities that screened out.

Appendix A Examples

The examples listed below are real-world examples and are presented in the format used at the facility that did the analysis.

Commented [HD58]: Generally, these examples do not describe how they use the guidance of this document, but they should do so. The examples need additional edits also. However, the guidance also needs significant work and should be finalized before attempting examples.

Commented [CG59]: So, all good practice examples should first be a screen, then an evaluation (as necessary for some). None of these provide that. Good practice examples should use a standard screening and evaluation format, so would help to have template forms for each to provide that standardization.

Commented [HD60R59]: The issue is that TRTRs format differs from 96-07, in which, each question was selected to illustrate the particular question.

TRTRs examples span multiple questions and considerations, BUT do not appear to follow the guidance provided herein.;

1. **Replacement of a process control cabinet with a digital process control system (PCS)**

The process control cabinet (PCC) serves as the control room assembly point for the location of the ventilation valve position indicators, core flow indicator, temperature recorder, flow recorder, conductivity recorder, fourteen motor pushbutton indicating controls, instrument power supply, and circuit breakers. One circuit breaker controls the instrument power to the process control cabinet and the other circuit breaker controls power to the pneumatic tube system.

Manual control of reactor cooling systems, water conditioning systems, containment building ventilation system, and the monitoring of these systems is accomplished via the PCC. Each of a number of process motors is controlled by a set of remote RUN and STOP indicating pushbuttons on the PCC panel. These motors include the primary and secondary pumps, secondary cooling tower fans, all ventilation exhaust fans, the makeup pump and the cleanup pump. The PCC also houses the chart recorders for the primary and secondary system flow rates, three primary system temperature measurements, and three water conductivity measurements.

There are no changes to Technical Specifications needed for this change. Of the scenarios analyzed in the FSAR, only Loss of Primary Flow could be considered as affected by changes to the process instrumentation and control system. Thus, the change *Screens In* to a 50.59 evaluation.

5.1.1 50.59 evaluation for PCS Replacement

1) Will the PCS installation and use result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?

No.

The frequency of occurrence of accidents is not analyzed in the FSAR and probabilistic risk assessments are not required for non-power reactors. However, a qualitative estimate may be provided for this evaluation. While the mechanism of the initiating events are not ~~presented~~ described or analyzed in the FSAR, the effect of the event is described. For example, a loss of coolant flow accident may be described in the FSAR, but the mechanism (i.e., pump stoppage) may not be described. The loss of flow may occur from the most likely mechanisms: loss of electrical power, a pump or motor failure, a controls failure (e.g., short circuit), or operator error ~~(i.~~e.g., switching off the pump). The first two mechanisms are unaffected by the change. ~~The probability of a control failure for either hard-wired pushbutton control or the new digital control is impossible to estimate since reliability (PRA) data is not available for either.~~ As noted in the PCC description above, the pushbutton components are nearly 30 years old and problems with the pushbuttons were becoming apparent. The probability of operator error is expected to be less for two reasons: (1) the loss of pushbutton indicators of pump on/off status on the PCC could lead to inadvertent operator error, (2) the digital control for primary pump shutoff on the PCS will have two step process whereby the operator selects the off control then must confirm the off command to shut down the pump.

2) Will the PCS installation and use 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 FSAR?

No.

The likelihood of occurrence of SSC malfunctions is not evaluated in the FSAR. Since probability risk assessments are not performed, there is no reliability data available on SSCs to adequately determine the probability of occurrence of malfunctions. As discussed above, these are the channels associated with the process variables of power, coolant temperature, coolant flow, and pool height, all of which are associated with the technical specifications safety limits and limiting safety system settings. These channels and their functionality remain unchanged. Unused or auxiliary analog output points from these devices are attached to the PCS using optically isolated components. The use of these outputs will not increase the likelihood of occurrence of a malfunction of these components. The addition of a dry contact scram relay in series with the scram relay chain (safety chain) does not negatively affect the safety chain. Any single open relay in the series circuit will result in a scram. The addition of another trip point provides additional diversity and redundancy to the reactor safety system. The integration of the PCS hardware and software has been functionally tested using the appropriate test procedures.

3) Will the PCS installation and use result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR?

No.

The FSAR analysis for coolant flow loss shows there is no consequence for the loss of coolant flow with the reactor at full power and reactor scram occurring at 80% the nominal flow rate. The change will have no effect on the consequence or lack thereof.

4) Will the PCS installation and use result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR?

No.

Malfunctions of SSC resulting in a radiological consequence are not analyzed in the FSAR. The consequences of an SSC malfunction would be bounded by the accident consequences evaluated in the FSAR. Except for the MHA, all the accident analyses show there is no consequence.

5) Will the PCS installation and use create a possibility for an accident of a different type than any previously evaluated in the FSAR?

No.

All credible accident scenarios have been evaluated in the FSAR. There are no credible accidents involving the PCS that would not be bounded by the accident scenarios in the FSAR.

6) Will the PCS installation and use create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR?
No.
The results of the accident scenarios analyzed in the FSAR, with the exception of the MHA, indicate there are no consequences. As provided previously in this evaluation, the only malfunction common to both the old and new system would be a malfunction resulting in a loss of coolant flow. The results of this malfunction (i.e., no consequence) would be the same.

7) Will the PCS installation and use result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?
No.
As analyzed in the FSAR, there are no reactor transients or credible accidents that pose a significant risk of fuel cladding failure. The design basis limit for the primary fission product barrier (fuel cladding) is the onset of nucleate boiling (ONB). The temperature for ONB is well below that for fuel cladding failure and the Technical Specifications limiting safety system settings (LSSS) assure that the ONB is not reached. As indicated previously in this evaluation, SSC associated with the LSSS will be unaffected by this change.

8) Will the PCS installation and use result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?
No.
The PCS installation does not involve a change to any analytical method used in the safety analyses provided in the FSAR.

DETERMINATION CONCLUSION:
The installation and operation of the PCS does not involve a modification of the Technical Specifications and meets the criteria specified in 10 CFR 50.59. As such, it may be implemented without prior NRC approval in accordance with 10 CFR 50.59.

2. Screening of Replacement of wide range power monitor

The overall approach to measuring and displaying power between the old and new channels is the same. What follows is a brief discussion of the differences and what the expected impact is.

The new preamplifier is located above the reactor top grating. This is a better environment from the standpoint of temperature and humidity. With the added volume in the new enclosure, more of the processing of the fission chamber signal is performed closer to the detector. This also eliminates the need to convey the detector high voltage bias from the console. All of this should lessen the impact of electrical noise.

The old log channel sums the count rate and MSV signals. The new channel switches between count rate and MSV and their respective period channels as well. There has always been some distortion as the contribution of count rate rolled off and MSV came on. This was most visible on the period meter. With switching instead of summing and two different period circuits, this crossover is expected to be much more abrupt. With the elimination of the period scram (Screen 19-07), it should have no serious impact on operation.

The existing linear channel uses 22 reed relays to switch around the signal for the 19 ranges. The new channel uses nine solid state switches to manage the signal for the nine ranges. This should be more reliable and easier to maintain due to the reduced complexity as well as easier-to-obtain spare parts.

The use of two staggered MSV circuits in the wide range linear channel means that the use of the count rate circuit stops at 10W instead of 100W. Using the high gain bandpass circuit for the middle ranges should improve the linearity of the channel overall.

1. Does the proposed activity involve a change to a SSC that adversely affects an FSAR (as updated) described design function? (See Section 4.2.1 of NEI 96-07)

No. Per the FSAR:

"The fission chamber is connected to a pre-amplifier at the reactor top. The pre-amplifier monitors the high voltage to the fission chamber, provides an input point for test signals, and preamplifies the fission chamber signal for use in the wide range log and linear channels. If a loss of high voltage to the fission chamber is sensed, a bistable circuit will be tripped, resulting in a scram.

The wide range log channel provides a continuous indication from 10^{-8} to 110% of full power for the local meter and the console data recorder. The log power level signal is also used by the low source count bistable circuit, the 1 kW permissive bistable, the pulsing channel, and the period channel.

The period signal is obtained by differentiating the wide range log signal. Reactor period is displayed on a local meter. A bistable circuit provides a scram and an alarm when rate exceeds a predetermined limit. The period signal is also used by the servo system. The period circuit is disabled initially in Square-Wave mode and at all times in Pulse mode.

The wide range linear channel provides a signal based on 0 to 110% of the range selected by a 19-position range switch. The wide range linear signal is used by the console data recorder and the servo system."

The new pre-amplifier will functionally be the same as the previous pre-amplifier. The new location of the pre-amplifier should improve its long-term availability. The new channel provides continuous indication from 10^{-8} to 110% on the local meter and console recorder and its signal is still used by the aforementioned circuits/channels. The new range switch is a 9-position switch and replaces an obsolete piece of equipment that has no spare.

Commented [CG61]: Was the original location (assumed BELOW top grating) cited in the FSAR as beneficial for other reasons or conceived advantage? Is there supporting data to support that the new location will actually lessen electrical noise?

Commented [HD62R61]: All technical positions should be supported by the engineering change package either by an incorporated analysis or reference to other documentation such that a third party can reasonably reach the same conclusion.

Commented [HD63R61]: Guidance to this effect should be provided within this document.

Commented [HD64]: Define acronym

Commented [HD65]: The statement that the period scram was apparently "Screened out" is questionable. Use of this statement should be with great caution and further explanation because it could in and of itself be misleading as guidance!

Commented [HD66R65]: Edit to be more generic as in "prior evaluation."

Commented [HD67R65]: Examples in the guidance carry the same weight as the text in the body of the document.

Commented [CG68]: This represents a change in instrument operation (method of performing or controlling a function). [See NEI 96-07 definition of change]. The NRC staff, expect discussion on the relative impact of this change. The "should have no serious impact" is, in this case, should have more elaboration.

Commented [CG69]: The NRC staff would expect a more detailed discussed as this is a change to the method of performing or controlling a function.

Commented [HD70]: The term "should" implies a lack of confidence which will result in additional regulatory scrutiny.

Commented [CG71]: A discussion of reliability should be provided as part of discussion under question 1 (9 vs 19).

2. Does the proposed activity involve a change to a procedure that adversely affects how FSAR (as updated) SSC design functions are performed or controlled? (See Section 4.2.1.2 of NEI 96-07)

   No. Procedures for testing safety functions (source count interlock, 1 kW pulse interlock) will fundamentally remain the same.

3. Does the proposed activity involve revising or replacing an FSAR (as updated) described evaluation methodology that is used in establishing the design basis or used in the safety analysis? (See Section 4.2.1.3 of NEI 96-07)

   No evaluation methodologies are changing due to this channel upgrade.

4. Does the proposed activity involve a test or experiment not described in the FSAR (as updated), where an SSC is utilized or controlled in a manner that is outside the reference bounds of the design for that SSC or is inconsistent with analyses or descriptions in the FSAR (as updated)? (See Section 4.2.2 of NEI 96-07)

   No. The new safety channel will be controlled/monitored the same fundamental way as the current channel.

5. Does the proposed activity require a change to the Technical Specifications?

   No Technical Specification changes will be required for this activity.

Commented [HD72]: Questions 2, 3, 4, and 5 do not contain sufficient information to make the conclusion of no impact. A reasonable 3rd person would not necessarily reach the same conclusion from the limited information provided. Additionally, each question should stand-alone (or at least reference external information) rather than implying that the change description is sufficient. Rather than relying on the reader to derive the information from the change description, a complete description directly relevant to the particular question should be provided.

Conclusion

The proposed replacement of a wide range monitor *screens out* of a 50.59 evaluation.

3. Upgrade of a fuel element temperature channel

The staff proposes to upgrade the fuel element temperature channel to an Omega CN8PT process meter.

The Technical Specifications require that fuel element temperature be monitored during operation (*Technical Specification 3.2.2*), and that a reactor scram occur if the Limiting Safety System Setting (LSSS, *Technical Specification 2.2*) of 510°C is exceeded (*Technical Specification 3.2.3*). The time from the fuel element temperature exceeding the LSSS to the slowest scrammable rod reaching its bottom limit shall not exceed 2 seconds (*Technical Specification 3.2.1*).

1. Would the proposed change, test or experiment result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR (as updated)? ☐YES ☒NO.

The proposed change, a new fuel temperature channel, will not result in an increase in the frequency of occurrence of an FSAR-evaluated accident. The nine credible accident scenarios for experimental reactors identified in NUREG 1537 are as follows:

a) Maximum Hypothetical Accident (MHA)
b) Insertion of Excess Reactivity
c) Loss of Coolant Accident (LOCA)
d) Loss of Coolant Flow
e) Mishandling/Malfunction of Fuel
f) Experiment Malfunction
g) Loss of Normal Electrical Power
h) External Events
i) Mishandling or Malfunction of Equipment

All FSAR-evaluated accidents have been extremely conservatively analyzed with the assumption that the reactor has been operating continuously for 1 year at full power. The reactor only operates at most for 12 hours in a day. The fuel will never approach the conditions required for the maximum hypothetical accident. There are no evaluated accidents that involve loss of fuel temperature indication. Historically, the IFE has never exceeded the LSSS setting of 510°C, even under maximum allowed pulsing conditions, so it is highly unlikely that the IFE would exceed the LSSS under normal operation conditions with a "fixed" failed temperature channel as the redundant safety and percent power channels would cause an overpower scram far before reaching the LSSS. A "fixed" indication would likely be discovered during hourly logs, as the channel is switched by the operator in order to log the three thermocouple readings. A high temperature channel failure would cause a scram, a low temperature failure would cause an annunciator alarm which would alert the operator.

2. Would the proposed change, test or experiment 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 FSAR (as updated)? ☐YES ☒NO.

From "Interim Staff Guidance Augmenting NUREG-1537, Part 1 (ADAMS Accession No. ML12156A069)," some potentially adverse effects of digital upgrades that should be evaluated include:

• *Replacing analog with digital equipment*

– software common-cause failures cannot be assumed to be incredible failures

The only possible software (which in this case is firmware) common-cause failure would be due to the channel incorporating both high and low alarms into the channel rather than having physically separate comparators, although the alarms are assigned to different memory locations. In the event of a channel failure, the reactor will be automatically scrammed. The firmware is password-protected and not remotely accessible.

– a digital system can fail "fixed" without giving any indication that it has failed

If the new channel were to fail "fixed", which would be a frozen reading, the reactor would scram due to other redundant safety features, particularly exceeding the safety power or percent power scram setpoints, which are 6% above normal

power. Normal IFE temperature would have to increase by over 10% to exceed the LSSS; this is not expected to happen at 6% above normal power. Historically, the IFE has never exceeded the LSSS setting of 510°C, even under maximum allowed pulsing conditions, so it is highly unlikely that the IFE would exceed the LSSS under normal operation conditions with a "fixed" failed temperature channel as the redundant power channels would cause an overpower scram far before reaching the LSSS. A "fixed" indication would likely be discovered during hourly logs, as the channel is switched by the operator in order to log the three thermocouple readings.

– a watchdog timer may add diversity and redundancy but does add a new failure mode
There is no watchdog timer.

• *Combining previously separate functions into one digital device such that failures create new malfunctions (i.e., multiple functions are disabled if the digital device fails)*
The previous channel utilized two separate thumbwheels to set the high and low temperature alarm setpoints. The new channel will effectively combine these alarms into the firmware. If the new channel fails in a frozen state, the operator will likely discover this during the hourly logs, as the channel is switched by the operator in order to log the three thermocouple readings. If the new channel fails high, it will cause a reactor scram. If it fails low, it will cause a rod withdrawal prohibit and an annunciator alarm, which will alert the operator.

• *Changing performance from SAR-described requirements (e.g., for response time, accuracy, etc.)*
Per the OSTR SAR Section 7.2.3.2 Temperature Measurements:

"A multipoint selector on the console allows selection of one of the thermocouples or a test signal for display and comparison against the high and low setpoints. The displayed temperature is also recorded on the console data recorder. The high and low setpoint comparators both send alarms to the annunciator panel. In addition, the high comparator sends a scram signal to the reactor protection system. The multipoint selector also allows test inputs to be selected. When the multipoint selector is not in a position corresponding to an active thermocouple, the rod withdrawal prohibit interlock is activated."

The multipoint selector on the new channel will be functionally the same as the current channel. It will have five channels to select from: the three IFE thermocouples, and two inactive test channels. The new channel will continue to display temperature on the console data recorder. Both the high and low setpoints will send alarms to the annunciator panel. The high temperature alarm will send a scram signal to the reactor protection system. The multipoint selector will allow two test inputs to be selected. And when the multipoint selector is not selected to one of the three active thermocouples, a rod withdrawal prohibit interlock is activated. Thus, there are no expected changes to performance from SAR-described requirements.

• *Changing functionality in a way that increases complexity, potentially creating new malfunctions*
New functionalities involve color-changing LEDs, buttons for viewing/setting the LSSS setpoint, and the physical changes in the location of the meter/selector switch. The postulated new malfunctions would involve operator error due to misunderstanding how to operate the new equipment. Operators will all be trained on how to use the new meter before they can operate the reactor. The only conceivable operator malfunction would involve changing the LSSS setpoint. Whereas the current channel required little to no experience to change the LSSS setpoints via an easily accessible thumbwheel, now an operator must not only understand how to change the LSSS setpoint via the buttons on the front of the meter, but they must also have access to a password to do so. Only the operating staff will have access to this password. The new channel buttons are only expected to be manipulated in order to check the LSSS setpoint for daily scram checks, so inadvertent LSSS setpoint changes are not expected as the hardware will hardly be exercised.

• *Introducing different behavior or potential failure modes that could affect the design function*
Per OSTR SAR Section 7.2.1 Design Criteria:
The instrumentation and control system is designed to provide the following:

- complete information on the status of the reactor and reactor-related systems;
- a means for manually withdrawing or inserting control rods;
- automatic control of reactor power level;
- automatic scrams in response to over power, excessive rate of change of power, and high fuel temperature;
- automatic scrams in response to a loss of operability of the power measuring channels; and
  monitoring of radiation and airborne radioactivity levels.

To bullet point 1, the new channel will provide the same information (fuel temperature for three possible IFE thermocouples) as before. To bullet point 4, the new channel will automatically scram in response to high fuel temperature faster than the previous channel. Bullet points 2, 3 and 5 are not relevant as this meter is not involved with manual control rod operation or automatic control of reactor power level. Thus, there are no expected introductions of different behaviors or potential failure modes that could affect the design function.

• *Changes that fundamentally alter (replace) the existing means of performing or controlling design functions*
– replacement of automatic action by manual action (or vice versa)
There is no fundamental change to this channel in regards to automatic actions. Operators will manipulate the channel in the same fashion, by using the selector switch to change between different IFE thermocouples. The only expected fundamental change is the daily check of the LSSS setpoint before operation. The current channel had a visible setpoint thumbwheel that was able to be visually checked at all times. The new channel will require the operator to change screens via pushbuttons in order to check the LSSS setpoint. This will be performed every day before operation, same as before. Operators will be trained on how to use the meter before they operate the reactor.

– changes to the man-machine interface
The new channel is more user-friendly. It has larger LED size and the color-coding should alert the operator in conjunction with annunciation. The meter, selector switch and test pushbutton are physically different, so operators will be trained on how to use the meter before they operate the reactor.

– changing a valve from "locked closed" to "administratively closed"
This is not relevant.

• *HSI changes that could lead to potential adverse effects*
– *Changes to parameters monitored, decisions made, and actions taken in the control of plant equipment and systems during transients*
The same parameters (three possible IFE thermocouples) are monitored. Decisions that need to be made by operators and actions taken mostly involve hourly logging of temperature, which requires operation of the selector switch. This will be fundamentally the same but operators will need to be trained on the subtle differences of operation (new LED, new location of meter and selector switch/test pushbutton).

– *Changes that could affect the overall response time of the human/machine system (e.g., changes that increase operator burden)*
The new channel scrams the reactor faster (by approximately 0.3 seconds), thus improving overall response time of the human/machine system. The only change that could conceivably increase operator burden is not being able to see the LSSS setpoint without changing screens on the meter. The LSSS setpoint will be checked before daily operation and it is not anticipated to be changed during the day's operation.

– *Fundamental changes in data presentation (such as replacing an analog meter with a numeric display or a multipurpose video display unit (VDU) where access to the data requires operator interactions to display)*
Fundamentally, the data presentation is the same. It is still an LED meter. The new channel is more user-friendly. It has larger LED size and the color-coding should alert the operator in conjunction with annunciation. The only change that requires more operator interactions to display is having to change screens on the meter to observe the LSSS setpoint. The LSSS setpoint will be checked before daily operation and it is not anticipated to be changed during the day's operation.

– *Changes that create new potential failure modes in the interaction of operators with the system (e.g., new interrelationships or interdependencies of operator actions and plant response or new ways the operator assimilates plant status information)*
Since the channel is functionally the same, there are no postulated new potential failure modes due to operator interaction. If the operator accidentally leaves the meter on an inactive channel, there will still be a rod withdrawal prohibit alarm. There is less potential for exceeding the LSSS as a password is required to change the LSSS setpoint.

3. Would the proposed change, test or experiment result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR (as updated)? ☐YES ☒NO.

The proposed change, a new fuel temperature channel, will not result in an increase in the consequence of an FSAR-evaluated accident. All FSAR-evaluated accidents have been extremely conservatively analyzed with the assumption that the reactor has been operating continuously for 1 year at full power. The reactor only operates at most for 12 hours in a day. The fuel will never approach the conditions required for the maximum hypothetical accident. Changing the fuel temperature channel will not cause an increase in the consequences of the analyzed accidents as any postulated consequence of a fuel temperature channel failure are bounded by the maximum hypothetical accident.

4. Would the proposed change, test or experiment result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR (as updated)? ☐YES ☒NO.

The proposed change, a new fuel temperature channel, will not result in an increase in the consequence of a malfunction of the fuel temperature channel. No FSAR-evaluated accidents consider a malfunction of the fuel temperature channel as they have been extremely conservatively analyzed with the assumption that the reactor has been operating continuously for 1 year at full power. The reactor only operates at most for 12 hours in a day. The fuel will never approach the conditions required for the maximum hypothetical accident. A malfunction of the new fuel temperature channel will not cause an increase in the consequences of a fuel temperature channel failure as it is bounded by the maximum hypothetical accident.

5. Would the proposed change, test or experiment create a possibility of an accident of a different type than any previously evaluated in the FSAR (as updated)? ☐YES ☒NO.

The proposed change, a new fuel temperature channel, will not create a possibility of an accident of a different type previously evaluated in the FSAR. The nine credible accident scenarios for experimental reactors identified in NUREG 1537 are as follows:

a) Maximum Hypothetical Accident (MHA)
b) Insertion of Excess Reactivity
c) Loss of Coolant Accident (LOCA)
d) Loss of Coolant Flow
e) Mishandling/Malfunction of Fuel
f) Experiment Malfunction
g) Loss of Normal Electrical Power
h) External Events
i) Mishandling or Malfunction of Equipment

A fuel element temperature failure would fall under accident i). Per the FSAR Section 13.2.9 "Mishandling and Malfunction of Equipment":

"No credible accident initiating events were identified for this accident class. Situations involving an operator error at the reactor controls, a malfunction or loss of safety-related instruments or controls, and an electrical fault in the control rod system were anticipated at the reactor design stage. As a result, many safety features, such as control system interlocks and automatic reactor shutdown circuits, were designed into the overall TRIGA® Control System (Chapter 7). TRIGA® fuel also incorporates a number of safety features (Chapter 4) which, together with the features designed into the control system, assure safe reactor response, including in some cases reactor shutdown."

The new fuel temperature channel will not change any interlocks or automatic reactor shutdown circuits.

6. Would the proposed change, test or experiment create a possibility of a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR (as updated)? ☐YES ☒NO.

Per NEI 96-07:

"Malfunctions of SSCs are generally postulated as potential single failures to evaluate plant performance with the focus being on the result of the malfunction rather than the cause or type of malfunction. A malfunction that involves an initiator or failure whose effects are not bounded by those explicitly described in the UFSAR is a malfunction with a different result. A new failure mechanism is not a malfunction with a different result if the result or effect is the same as, or is bounded by, that previously evaluated in the UFSAR."

A failure of the new temperature channel would be bounded by the previously-evaluated accidents. All FSAR-evaluated accidents have been extremely conservatively analyzed with the assumption that the reactor has been operating continuously for 1 year at full power. The reactor only operates at most for 12 hours in a day. The fuel will never approach the conditions required for the maximum hypothetical accident, thus a failure of the new temperature channel would be bounded by the previously-evaluated accident just as the current temperature channel is bounded.

7. Would the proposed change, test or experiment result in a design basis limit for a fission product barrier as described in the FSAR (as updated) being exceeded or altered? ☐YES ☒NO.

The proposed change, a new fuel temperature channel, will not result in a design basis limit for an FSAR-described fission product barrier being exceeded or altered. The Technical Specifications require the LSSS to be set at 510 °C. This setpoint will not change. The new channel's LSSS is more secure. The current channel utilizes easily-accessible thumbwheels that anyone could easily manipulate. The new channel requires a password to access the setpoints, a password that is only available to the operating staff. The new channel also requires an understanding of how to change the setpoint, which is not obvious.

8. Would the proposed change, test or experiment result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design basis or in the safety analysis? ☐YES ☒NO.

The proposed change, a new fuel temperature channel, will not change the method of evaluation used to establish the design basis or safety analysis described in the FSAR. Per OSTR SAR Section 7.2.1 Design Criteria:

The instrumentation and control system is designed to provide the following:

- complete information on the status of the reactor and reactor-related systems;
- a means for manually withdrawing or inserting control rods;
- automatic control of reactor power level;
- automatic scrams in response to over power, excessive rate of change of power, and high fuel temperature;
- automatic scrams in response to a loss of operability of the power measuring channels; and monitoring of radiation and airborne radioactivity levels.

To bullet point 1, the new channel will provide the same information (fuel temperature for three possible IFE thermocouples) as before. To bullet point 4, the new channel will automatically scram in response to high fuel temperature faster than the previous channel. Bullet points 2, 3 and 5 are not relevant as this meter is not involved with manual control rod operation or automatic control of reactor power level. Thus, there are no changes to the design basis.

4. Replacement of Water/Air Temperature Console Equipment with PLC and Touchscreen HMI

The staff proposes to install new water/air temperature equipment in the control room console.

The current water/air temperature console equipment consists of two Fluke 2170A digital thermometers, an Omega digital comparator, an Omega ten point selector switch (all of which are in the right hand drawer), and five pushbuttons on the center control console that are used to turn on/off the cooling tower fans #1/#2, secondary pump, primary pump, and demineralizer pump.

One of the Fluke digital thermometers is used solely for "Bulk Water No. 2 Temperature" which displays the temperature from the #2 bulk water thermocouple in the reactor tank. The digital comparator is set to give an annunciator alarm when the "Bulk Water No. 2 Temperature" is above its thumbwheel setting (typically at 42°C). This gives the operator ample time to diagnose whether the reactor needs to be shut down before reaching the Technical Specification limit of 49°C (*Technical Specification 3.3.b*).

The other Fluke digital thermometer is capable of singularly displaying the temperature from ten different thermocouples: 1) Bulk Water No. 1 Temperature, 2) Reactor Tank Inlet Temperature, 3) Reactor Tank Outlet Temperature, 4) Primary Water Inlet to Heat Exchanger, 5) Primary Water Outlet from Heat Exchanger, 6) Demineralizer Bed Temperature, 7) Secondary Water Inlet to Heat Exchanger, 8) Secondary Water Outlet from Heat Exchanger, 9) Outside Air Temperature and 10) Console Air Temperature.

The proposed new water/air temperature equipment consists of a programmable logic controller (PLC) and touchscreen human-machine interface (HMI). The five aforementioned pushbuttons on the center console will be deactivated and the primary, secondary, and demineralized water pumps and the cooling tower fan will be controlled by tactile pushbuttons on the side of the new HMI. A pushbutton, independent of the HMI, will be provided for simultaneous shutdown of all three pumps and the cooling tower fan, which will be advantageous in the case of emergency shutdown of cooling.

1. Would the proposed change, test or experiment result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR (as updated)? ☐YES ☒NO.

The proposed change, an upgrade to the water/air temperature measuring channels, will not result in an increase in the frequency of occurrence of an FSAR-evaluated accident. The nine credible accident scenarios for experimental reactors identified in NUREG 1537 are as follows:

a) Maximum Hypothetical Accident (MHA)

b) Insertion of Excess Reactivity

c) Loss of Coolant Accident (LOCA)

d) Loss of Coolant Flow

e) Mishandling/Malfunction of Fuel

f) Experiment Malfunction

g) Loss of Normal Electrical Power

h) External Events

i) Mishandling or Malfunction of Equipment

The relevant analyzed accident would be OSTR SAR Section 13.2.4.2.1 "Loss of Coolant Flow Without Immediate Operator Action":

"If the [reactor] were operated without coolant flow for an extended period of time, and there was no heat removal by the reactor coolant systems, voiding of the water in the core could occur and the water level in the reactor tank would decrease because of evaporation. The sequence of events postulated for this very unlikely scenario is as follows:

- the reactor would continue to operate at a power level of 1 MW (provided the rods were adjusted to maintain power) and would heat the tank water at a rate of about 0.82°C/min until the tank water reached the bulk water high temperature alarm setpoint. This setpoint is at 42°C. The normal bulk water temperature, when operating at 1 MW, ranges from 35 to 40°C, depending on the outside air temperature. Thus, it would take from 2.4 minutes to 8.5 minutes to reach the alarm setpoint. Assuming the operator did not notice this alarm and did not take any corrective action, the bulk water would continue to rise above 42°C. It would then take an additional 70.7 minutes for the water in the tank to reach the saturation temperature. At this time, voids in the core would cause power oscillations and the negative void coefficient would cause a reduction in power if control rods were not adjusted to maintain power; and
- if it is assumed that the operator or automatic control system maintained power at 1 MW, and still assuming that the system is adiabatic except for the evaporation process, about 1,596 kg/hr would be vaporized. The reactor tank water level would decrease, and it would take about 9.4 hours for the water level to reach the top of the core, and an additional 1.5 hours to vaporize all of the water remaining in the tank. The reactor, however, would shut down as the water level dropped past the top of the fuel.

It is considered inconceivable that such an operating condition would go undetected. Water level, water flow, and water temperature alarms would certainly alert the operator. Also, as the water level decreases, the reactor room radiation monitors would alarm. Because of all of these factors, water would be added to the tank and/or the reactor would be shut down to mitigate the problem."

The new equipment will continue to alarm at 42°C but is more conservative as now both reactor tank thermocouples will cause annunciation, as opposed to only "Bulk Water No. 2". This accident assumes the operator did not notice the alarm, so extreme conservatism has already been analyzed. The frequency of occurrence of this accident does not increase more than a minimum as a result of this equipment upgrade.

2. Would the proposed change, test or experiment 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 FSAR (as updated)? ☐YES ☒NO.

From "Interim Staff Guidance Augmenting NUREG-1537, Part 1 (ADAMS Accession No. ML12156A069)," some potentially adverse effects of digital upgrades that should be evaluated include:

• *Replacing analog with digital equipment*

– software common-cause failures cannot be assumed to be incredible failures

The only possible software (which in this case is firmware) common-cause failure would be due to the HMI incorporating multiple alarms and controls. This should not matter as this system is not safety-related. The worst possible loss-of-coolant accident takes over nine hours until possible uncovering of the core.

– a digital system can fail "fixed" without giving any indication that it has failed

If the new channel were to fail "fixed", which would be a frozen reading, it would likely be discovered during hourly logs. The likelihood of all thermocouples having the same consecutive log readings is infinitesimal. As the worst possible loss-of-coolant accident takes over nine hours to uncover the core, the "fixed" indication would be discovered far before this accident could happen.

– a watchdog timer may add diversity and redundancy but does add a new failure mode

There is no watchdog timer.

• *Combining previously separate functions into one digital device such that failures create new malfunctions (i.e., multiple functions are disabled if the digital device fails)*

The previous channel utilized a thumbwheel to set the high reactor water temperature alarm setpoint, and a separate digital meter that annunciated a reactor water low temperature alarm. The new channel will effectively combine these alarms into the firmware. If the new channel fails in a frozen state, the operator will likely discover this during the hourly logs. If the new channel fails low or high, it will cause an annunciator alarm which will alert the operator.

• *Changing performance from SAR-described requirements (e.g., for response time, accuracy, etc.)*

Per the SAR Section 7.2.3.2 Temperature Measurements:

"Temperature of the bulk pool water is measured by two thermocouples located on either side of the primary tank. One thermocouple is connected to a dedicated thermometer on the console for display. A setpoint comparator sends a high temperature alarm to the annunciator panel when the measured temperature exceeds the setpoint. A selector switch allows the second bulk water thermocouple as well as various points throughout the primary, secondary, and demineralized water systems to be selected for display on the console."

The temperature of the pool water is still measured by the same two thermocouples. They will now both be connected to a dedicated thermometer at all times (on the HMI). They will both send a high temperature alarm to the annunciator panel. There will no longer be a selector switch as all points will be on constant display on the graphical interface. Thus, there are no expected adverse changes to performance from SAR-described requirements.

• *Changing functionality in a way that increases complexity, potentially creating new malfunctions*

The postulated new malfunctions would involve operator error due to misunderstanding how to operate the new equipment. Operators will all be trained on how to use the new equipment before they can operate the reactor. The new equipment should functionally be similar with regards to how they are operated (pushbuttons) and will require less operator action as the new equipment will automatically control temperature, whereas the old system required manual cycling of fan pushbuttons.

• *Introducing different behavior or potential failure modes that could affect the design function*

Per SAR Section 7.2.1 Design Criteria:

"The instrumentation and control system is designed to provide the following:
- complete information on the status of the reactor and reactor-related systems;
- a means for manually withdrawing or inserting control rods;
- automatic control of reactor power level;
- automatic scrams in response to over power, excessive rate of change of power, and high fuel temperature;
- automatic scrams in response to a loss of operability of the power measuring channels; and monitoring of radiation and airborne radioactivity levels."

Bullet points 2 through 5 are not relevant. To bullet point 1, the new channel will provide the same temperature information as before, in addition to a reactor top ambient air temperature thermocouple. All temperatures will be displayed on one graphical screen, thus providing better information on status. Thus, there are no expected introductions of different behaviors or potential failure modes that could affect the design function.

- *Changes that fundamentally alter (replace) the existing means of performing or controlling design functions*

  – replacement of automatic action by manual action (or vice versa)
  The new cooling system will have automatic temperature control, as opposed to the current system which is manually controlled by cycling two cooling tower fans off and on to respond to cooling demand. The new system utilizes one cooling tower fan that will vary its speed according to cooling demand.

  – changes to the man-machine interface
  The meter and pushbuttons are physically different, so operators will be trained on how to use the meter before they operate the reactor. The HMI will have a graphical display that should give more intuitive information than a singular meter connected to a rotating 10-point selector switch.

  – changing a valve from "locked closed" to "administratively closed"
  This is not relevant.

- *HSI changes that could lead to potential adverse effects*

  – Changes to parameters monitored, decisions made, and actions taken in the control of plant equipment and systems during transients
  The same parameters are monitored, in addition to reactor top ambient air temperature. Decisions that need to be made by operators and actions taken mostly involve hourly logging of temperature. There are less operational decisions to be made as the cooling will now be automated. The hourly logging will fundamentally be the same but operators will need to be trained on the subtle differences of operation (new location of meter, new temperature display and pushbuttons). Equipment can be easily shutdown in cases of emergency with a single pushbutton.

  – Changes that could affect the overall response time of the human/machine system (e.g., changes that increase operator burden)
  The new channel should not increase operator burden and should decrease response time. All relevant information will be displayed on the HMI. The pushbuttons for operating equipment will be directly on the HMI as opposed to being on another part of the console as the current system is. Equipment can be easily shutdown in cases of emergency with a single pushbutton.

  – Fundamental changes in data presentation (such as replacing an analog meter with a numeric display or a multipurpose video display unit (VDU) where access to the data requires operator interactions to display)
  Fundamentally, the data presentation is the same, but instead of having to operate a selector switch to log temperatures, all of the temperatures will be simultaneously displayed on the new channel. All relevant cooling controls and information will be on the HMI as opposed to three separate panels.

  – Changes that create new potential failure modes in the interaction of operators with the system (e.g., new interrelationships or interdependencies of operator actions and plant response or new ways the operator assimilates plant status information)
  Since the channel is functionally the same, the only postulated new potential failure modes would be due to an operator misunderstanding how to operate the new equipment. Operators will all be trained on how to use the new equipment before they can operate the reactor.

3. Would the proposed change, test or experiment result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR (as updated)? ☐YES ☒NO.

The proposed change, an upgrade to the water/air temperature measuring channels, will not result in more than a minimal increase in the consequences of any FSAR-evaluated accident. The worst-case scenario is no operator action on loss of coolant, and even with that highly unlikely scenario, it would still take 9 hours to uncover the reactor core. The new temperature equipment will not increase the likelihood of the accident, nor the consequences of this accident.

4. Would the proposed change, test or experiment result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR (as updated)? ☐YES ☒NO.

The proposed change, an upgrade to the water/air temperature measuring channels, will not result in an increase in the consequences of a malfunction of an SSC. The worst-case scenario is no operator action on loss of coolant, and even with that highly unlikely scenario, it would still take 9 hours to uncover the reactor core. The new temperature equipment will not increase the consequences of a malfunction of any SSCs important to safety.

5. Would the proposed change, test or experiment create a possibility of an accident of a different type than any previously evaluated in the FSAR (as updated)? ☐YES ☒NO.

The proposed change, an upgrade to the water/air temperature measuring channels, will not create a possibility of a new type of accident. There has already been an analysis of a loss of coolant accident without operator action (OSTR SAR Section 13.2.4.2.1).

6. Would the proposed change, test or experiment create a possibility of a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR (as updated)? ☐YES ☒NO.

The proposed change, an upgrade to the water/air temperature measuring channels, will not create a possibility of a new type of SSC malfunction. There has already been an analysis of a loss of coolant accident without operator action (OSTR SAR Section 13.2.4.2.1), which would encompass any malfunction of the new temperature metering equipment.

7. Would the proposed change, test or experiment result in a design basis limit for a fission product barrier as described in the FSAR (as updated) being exceeded or altered? ☐YES ☒NO.

The proposed change, an upgrade to the water/air temperature measuring channels, will not affect the design basis limit for the fission product barrier, which is the fuel cladding. The new temperature metering equipment has no direct effect on fuel temperature and the worst-case failure of the new metering equipment would take 9 hours to affect the fuel cladding, exactly the same as the current metering equipment.

8. Would the proposed change, test or experiment result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design basis or in the safety analysis? ☐YES ☒NO.

Per the SAR Section 7.2.3.2 Temperature Measurements:

"Temperature of the bulk pool water is measured by two thermocouples located on either side of the primary tank. One thermocouple is connected to a dedicated thermometer on the console for display. A setpoint comparator sends a high temperature alarm to the annunciator panel when the measured temperature exceeds the setpoint. A selector switch allows the second bulk water thermocouple as well as various points throughout the primary, secondary, and demineralized water systems to be selected for display on the console."

The temperature of the pool water is still measured by the same two thermocouples. The evaluation has been improved for the operator as now both thermocouples are connected to a high temperature alarm as opposed to only one. Also, all temperatures will be simultaneously visible on a graphical interface, as opposed to having to utilize a selector switch to view specific temperatures.

5. Replacement of Control Room Alarm Annunciator

The control room annunciator panel is being replaced. The existing panel is the original installed in 1974. The original alarm panel has 17 individual alarm indicators (2"x3" each), each illuminated by two 120VAC incandescent light bulbs. The new panel will have indicators (2.25"x2.75" each), each illuminated by multiple LEDs. 6 additional alarm indicators have added to provide additional information for the operator.

Screening questions:

1. Does the proposed activity adversely affect an SSC design function described in the FSAR?

   No. The original alarm was panel was designed to provide a visible and audible annunciation of an alarm condition to the operator. The new alarm panel will perform the same function. The audible indicator (alarm buzzer) is not being changed, only the visible alarm indicators. The new indicators will use more reliable LED technology rather than incandescent lights. The indicators are slightly larger and will be in the same physical location in the control room.

2. Does the proposed activity adversely affect a method of performing or controlling an SSC design function described in the FSAR?
   No. The audible and visible alarm indicators are triggered by relays and switches. For example, the "Short Period" alarm is triggered by a relay in the Logarithmic Power/Period module. The alarm triggers are not being changed, only the indicators.
3. Does the proposed activity involve revising or replacing an evaluation methodology that is used in establishing a design bases or used in a safety analyses described in the FSAR?
   No. The proposed activity is not associated with or applicable to evaluation methodologies in the FSAR.
4. Does the proposed activity involve a test or experiment not described in the FSAR, where an SSC is used or controlled in a manner that is outside the reference bounds of the design for that SSC, or is inconsistent with analyses or descriptions presented in the FSAR?
   No. The change does not involve a test or experiment as defined for 50.59 evaluation purposes.

Conclusion: The annunciator panel replacement *screens out* of a 50.59 evaluation.

Commented [HD73]: The example only contains four questions and they do not appear to align with the guidance on screening provided in this document. If this example is to used, all seven questions should be represented.

Commented [CG74]: There is no discussion of additional alarm indicators and the 'additional" information provided, including no discussion of potential for mis-diagnosis or contradictory indication for the range of alarm indication provided in comparison to function as described in FSAR.

Commented [CG75]: Example screening and evaluation forms should be included to provide standardization basis for licensees.