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Thomas A. Marlow
Director
Nuclear Safety Assurance

0CAN050607

March 30, 2006

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: 10 CFR 50.59 Summary Report and Commitment Change Summary Report
Arkansas Nuclear One, Unit 1 and Unit 2
Docket No. 50-313 and 50-368
License No. DPR-51 and NPF-6

Dear Sir or Madam:

In accordance with 10 CFR 50.59(d)(2), enclosed is the Arkansas Nuclear One – Unit 1 (ANO-1) 10 CFR 50.59 summary report for the time period ending May 30, 2006. This report contains a brief description of changes in procedures, the facility as described in the ANO-1 Safety Analysis Report (SAR), changes in the ANO-1 Technical Requirements Manual (TRM), and changes in the ANO-1 Technical Specification (TS) Bases. The report also contains a description of tests and experiments conducted, if any, which were not described in the SAR, and other changes to the SAR for which a safety evaluation was conducted. A copy of each safety evaluation, both ANO-1 specific and those evaluations common between ANO-1 and ANO - Unit 2 (ANO-2), is included on the enclosed CD-ROM.

Included on the enclosed CD-ROM is the Commitment Change Summary Report for ANO-1 and ANO-2. These changes are submitted in accordance with the guidance provided in NEI 99-04, *Guidelines for Managing NRC Commitment Changes*. The report lists each commitment changed since submittal of the previous report and provides a basis for each change.

If you have any questions or require additional information, please contact David Bice at 479-858-5338.

Sincerely,

Thomas A. Marlow

Thomas A. Marlow

TAM/dbb

Enclosures (on single CD-ROM):

1. 10 CFR 50.59 Summary Report
2. ANO-1 and ANO-2 Commitment Change Summary Report

cc: Dr. Bruce S. Mallett
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NRC Senior Resident Inspector
Arkansas Nuclear One
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U. S. Nuclear Regulatory Commission
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Enclosure 1

To

0CAN050607

10 CFR 50.59 Summary Report

10 CFR 50.59 Summary Report

50.59 #	**50.59 Summary**
04-006	Engineering Request ER-ANO-2004-0011-000, "BWST (T-3) Silica Cleaning with Reverse Osmosis Unit"
04-021	Engineering Request ER-ANO-2002-1078-006, "Original OTSG Storage Facility"
04-025	Procedure Revision OP-1409.638, "ANO-1 Intake Structure Natural Convection Cooling Test"
04-029	Engineering Request ER-ANO-2002-1078-014, "ANO-1 SG/RVCH Replacement – 2 Inch & Under Piping Evaluation"
05-001	Engineering Request ER-ANO-2003-0063-000, "Addition of Zinc Injection System"
05-003	Temporary Alteration TAP 03-1-009, "Main Transformer ‘C’ Deluge Piping Removal"
05-005	Procedure Revision OP-1103.004, "Soluble Poison Concentration Control"
05-009	Engineering Request ER-ANO-2002-1078-021, "Reactor Building Temporary Power"
05-010	Engineering Request ER-ANO-2002-1078-007, "Reactor Building Construction Opening"
05-011	Engineering Request ER-ANO-2002-1078-011, "Rigging & Handling Inside the Reactor Building"
05-014	Engineering Request ER-ANO-2002-0366-000, "ANO-1 CRDM and Service Structure HVAC Ductwork Modifications"
05-015	Engineering Request ER-ANO-2002-0638-000, "ANO-1 Reactor Vessel Closure Head (RVCH) Replacement"
05-016	Engineering Request ER-ANO-2002-1078-015, "OTSG Removal & ROTSG Preparation and Installation"
05-017	Engineering Request ER-ANO-2002-1078-018, "SG/RVCH Replacement - Interference Removal and Replacement"
05-018	Engineering Request ER-ANO-2002-0639-000, "ANO-1 Service Structure Replacement"
05-020	Engineering Request ER-ANO-2000-2294-000, "E-4A/B & E-5A/B Feedwater Heater Replacement"
05-022	Engineering Request ER-ANO-2002-0640-000, "Polar Crane Up-Rates to 190 Tons"
05-023	Calculation CALC-ANO-ER-05-030, "Part 50 Analysis of an MPC-24 Dry Fuel Storage Cask for ANO-1"
05-025	Engineering Request ER-ANO-2002-1078-016, "SG/RVCH Replacement - Insulation"
05-026	Engineering Request ER-ANO-2002-1078-018, "SG/RVCH Replacement – Interference Removal and Replacement"
05-028	Engineering Request ER-ANO-2005-0481-000, "Removal of ECP Spillway Tarp"
05-029	Procedure Revision OP-1015.001, "RCP Seal Injection Control Valve Controller"
05-030	Engineering Request ER-ANO-2002-1381-000, "Replacement OTSG Project"
05-031	Engineering Request ER-ANO-2005-0566-002, "Main Chiller Cooling Water"

(continued)

50.59 #	50.59 Summary
05-032	Engineering Request ER-ANO-2004-0020-000, "CRDM Modifications"
05-033	Engineering Request ER-ANO-2005-0491-000, "ANO-1 Cycle 20 Reload Report and COLR"
05-034	Engineering Request ER-ANO-2002-0640-000, "ANO-1 Polar Crane Uprate"
06-001	Engineering Request ER-ANO-2004-0944-000, "Evaluation of Elevated Station Battery Specific Gravity"
06-002	Procedure Revision OP 1000.152, “Compensatory Actions for Reactor Building Cable Spreading Area Fire Detection System”

ANO 50.59 Evaluation Number

2004-006

I. OVERVIEW / SIGNATURES

Facility: ANO-1

Document Reviewed: *ER-ANO-2004-0619-000, OP-1604.053* **Change/Rev.:** 0
(FFN 04-006 originally reviewed ER-ANO-2004-0011-000)

System Designator(s)/Description: SFP

Description of Proposed Change:

The Engineering Evaluation and new procedure evaluated herein controls the installation of a Temporary Reverse Osmosis Unit for Silica Removal from ANO-1 Borated Water Storage Tank via a procedurally-controlled Temporary Alteration. Controls will be in place to monitor boron levels and chemistry parameters and to take certain mitigating actions to minimize the effects of pipe breaks or seismic activity. This 50.59 addresses installation and plant interface connections. The procedure contains instructions for staging, operating, and removing the Reverse Osmosis unit.

Check the applicable review(s): **(Only the sections indicated must be included in the Review.)**

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 04-006 Rev 1)**	**Sections I, II, and IV required**

Preparer: Robert J. Priore / **ORIGINAL SIGNED BY ROBERT J. PRIORE** / EOI / SYE / 08-23-05
Name (print) / Signature / Company / Department / Date

Reviewer: John Souto / **ORIGINAL SIGNED BY JOHN SOUTO** / EOI / SYE / 08-23-05
Name (print) / Signature / Company / Department / Date

OSRC: J. R. Eichenberger / **ORIGINAL SIGNED BY J. R. EICHENBERGER** / 09-15-05
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings and 50.59 Evaluations.]

II. SCREENINGS

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	***N/A***
TS	☐	☒	***N/A***
NRC Orders	☐	☒	***N/A***
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See LI-101 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	***3.2.4.3.2.A/J, fig 3-70, 4.3.2, 7.2.2.2.1, 7.2.2.3.1, 7.2.2.3.4, Figure 7-21***
TS Bases	☒	☐	***B.3.1.4***
Technical Requirements Manual	☐	☒	***N/A***
Core Operating Limits Report	☐	☒	***N/A***
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	***N/A***
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	***N/A***
If "YES," perform an Exemption Review per Section III OR perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the LBD change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	***N/A***
Emergency Plan[2, 3]	☐	☒	***N/A***
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	***N/A***
Offsite Dose Calculations Manual[3, 4]	☐	☒	***N/A***
If "YES," evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113.			

[1] If "YES," see LI-101. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☐ **Yes**
☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113, if applicable. If obtaining NRC approval, document the change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.

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3. Basis

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

I have reviewed the Engineering Evaluation of the Installation of a temporary Reverse Osmosis skid to increase the purification rate of the Spent Fuel Pool Purification System (also used to purify the BWST). I found one impact on the ***Final Safety Analysis Report*** if and when that configuration is installed in the plant. No other impacts were observed. The Engineering Evaluation and plant procedure allow for operation of the RO equipment in several different modes. The BWST is required to be operable in modes 1-4. If this equipment is utilized in modes 5 or 6 several requirements are relaxed.

Operating License

The ANO-1 Borated Water Storage Tank is referenced in the Operating License and Technical Specifications, however, the proposed change is below the level of detail and control found in these documents. Therefore, the Operating License will not be impacted by implementation of the configuration reviewed in this Engineering Evaluation.

Final Safety Analysis Report:

The FSAR does not state any limitations on using supplemental equipment to increase the purification rate of the Spent Fuel Pool Purification System. However, in consideration of the fact that another leakage path out of the Purification System is being introduced, **a 50.59 Evaluation will be performed.** The FSAR depicts valves SF-1032 and SF-1039 on figure 9-11. These valves will be manipulated to "valve in" the Reverse Osmosis unit. This configuration will support the use of a Reverse Osmosis skid that will remove silica, particulate matter, and hydrogen peroxide from the contents of the BWST. The installation, running, and removal of this unit will be controlled under a Procedurally-controlled temporary alteration that will reference ***ER-ANO-2004-0619-000***.

Other impacts on the plant reviewed for FSAR impacts:

Changing Water chemistry. The BWST water will be filtered and returned to the BWST when this configuration change is installed. The boron concentration of the returned water will be slightly lower than it was prior to treatment. This Engineering Evaluation contains work control steps that are to be integrated into controlled site documents (Work Plan, Procedures, etc.) to assure boron concentration requirements are adhered to while this equipment is in use.

Component Failure. The limiting Failure Mode for the Reverse Osmosis system would be a piping rupture with uncontrolled full flow influent discharge. The Engineering Evaluation contains instructions for countering this eventuality by maintaining BWST level high, posting a continuous firewatch that will also monitor for leaks, and providing operator guidance to secure the unit in the event of a seismic event, Safety Injection Actuation Signal, or piping rupture. Any rupture that occurs on the skid will be bounded by flow rates from Spent Fuel Pool purification piping rupture events. However, even though the flow rate out of a postulated break would be bounded so long as piping and hoses associated with this process are smaller than the smallest break evaluated, it appears that the likelihood of this type of accident will be increased very slightly. The equipment that the skid is constructed from is also very robust and constructed of high-quality equipment.

Creating Wastewater: Not all of the water fed to the skid is returned to the BWST. Some of the water is sent to the liquid rad waste system (LRW) for processing. The Engineering Evaluation confirms that the LRW can easily process the water generated by this process with no impact on normal operations.

Tests or Experiments Considerations:

The configuration reviewed by this evaluation is not a new Test or Experiment and will not require the BWST to be operated in modes for which it was not previously analyzed.

4. **References**

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
50.59 Unit 1	***Spent Fuel Pool, Borated Water Storage, BWST, silica, purification, osmosis, filter w/10 boron, Spent Fuel w/10 Chemistry***

LBDs/Documents reviewed manually:

FSAR Figure 9-11, 11-5, FSAR section 9.5.

5. **Is the validity of this Review dependent on any other change?**

☐ **Yes**

☒ **No**

If "YES," list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

No license changes are required to support this design change.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1]See NMM Procedure EV-117 for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following question if one of questions C.1 through C.10 above was answered "yes."

Is a change to the Security Plan required?

☐ Yes
☒ No

Attach to this 50.59 Review or reference below documentation for accepting a "yes" answer for any of Questions C.1 through C.10, above.

____

Name of Security Plan reviewer (print / Signature / Date)

D. **INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING (NOTE: This section is not applicable to Grand Gulf or Waterford 3 and may be removed from 50.59 Reviews performed for Waterford 3 proposed activities.)**

If any of the following questions is answered "YES," a 72.48 Review must be performed in accordance with NMM Procedure LI-112 and attached to this 50.59 Review.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function:

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists. Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof.
Reference: _____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions.

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.	☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The proposed change will not affect the overall performance or reliability of the Spent Fuel Pool Purification system. It will not cause the system to be operated outside of its design or test limits. The change will not affect the system's interface in any way that could lead to an accident, degrade any safety system or increase the possibility of an operator error. While this Reverse Osmosis system IS being connected to the purification system, it's connection will be controlled with an approved plant procedure. There will be component manipulations controlled in that plant procedure written to minimize the likelihood of operator errors. The SFP Recirculating system is not the initiator of any accidents identified in the FSAR. The pressure ratings of the RO unit are above the design pressure ratings of the SFP Recirculating System and therefore will not result in an increase in the frequency of occurrence of the Fuel Handling Accident described in Chapter 14 of the ANO-1 FSAR. There is likewise no apparent impact on the frequency of occurrence of faults described in chapter 9 of the ANO-1 FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   While the overall likelihood of a leak in the Spent Fuel Poor Purification system will increase slightly with the installation of this temporary configuration alteration, the requirement to be prepared to isolate the system in the event of a leak occurring on the skid or a seismic event occurring will offset this increase.

   This RO system is non-seismic and will be connected to a non-seismic portion of the SFP Purification system. Operation of this non-seismic system for long periods of time has previously been addressed in regards to continuous operability of the BWST. Plant procedures require that the non-seismic portions of the recirculation system be isolated in the event of failures induced by natural emergencies. Analysis has also been performed documenting that flow out of a pipe break in a non-seismic line will not render the BWST inoperable prior to isolation of the non-seismic line.

   If the Low Level BWST alarm comes in at the same time there is a pipe break on the unit there will be approximately 1 hour and 22 minutes for Operations to isolate the RO unit before the Tech Spec low limit is reached. There are steps in the procedure directing Operations to secure and isolate the RO unit before this level is reached.

   The Reverse Osmosis unit is constructed of high-quality commercial material and has been used repeatedly at ANO in a variety of applications with no history of failure. It is maintained properly by plant personnel as a tool and will be carefully checked out upon startup according to the controlling procedure. It is expected to perform well with no leaks or other non-conformances. It will also be monitored continuously by a firewatch who will, by procedure, be briefed on his/her responsibility to monitor the entire installation for leakage.

   Based on the above, there is less than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The accidents in Chapter 14 of the ANO-1 FSAR, including the fuel handling accident (the only accident explicitly referencing the Spent Fuel Pool or the Spent Fuel Pool Recirculating system) were evaluated for potential increases in dose consequences. The main accident mitigator in the ANO-1 Auxiliary Building is the Auxiliary Building Ventilation System and there will be no impact on this system with the use of this procedure. This change will not affect the source term or any radioactivity removal mechanism related to any of the accidents evaluated in the FSAR. Therefore, this change will have no impact on consequences of accidents evaluated in chapter 14 of the FASR nor will it result in any assumptions made in evaluating the consequences of an accident described in the FSAR invalid. In addition, the change will not affect the function of equipment designed to control the release of radioactive material nor will the change result in a new pathway of radioactive material to the environment.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

There will be no increase in the dose received by on-site personnel, control room operators or the public in the event that a structure, system, or component important to safety malfunctions after this change is implemented. The potential to have an uncontrolled release via the alteration hose connections and/or pipe break has been compensated for by Operation actions to ensure adequate initial BWST inventory to allow time for isolation. The change will not affect the function of equipment designed to control the release of radioactive material nor will the change result in a new pathway of radioactive material. If there was to be a failure of a hose from the change, this leak would be contained within the Unit 1 Auxiliary Building. Based upon this design feature of the Auxiliary Building and the operator instructions contained in the controlled plant document, there is less than a minimal increase in the consequences of any accident or to consequences due to any malfunction of a SSC important to safety.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

Several different scenarios of accidents related to operation of the associated equipment have been evaluated in support of the engineering review of the proposed procedure including possible seismic events, operator inattention, and equipment failures. The equipment and process proposed for installation will not create the possibility of any new types of accidents or equipment failure modes.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

Again, several different scenarios of accidents have been evaluated in support of the engineering review of the proposed procedure including possible seismic events, operator inattention, and equipment failures. If the Low-Level Alarm for the ANO-1 Borated Water Storage Tank (BWST) came in at the same time there was a Reverse Osmosis Unit hose break we have estimated that there would be approximately 1 hour and 22 minutes before the Tech Spec low BWST limit was reached. The controlling plant document will have steps in it to direct operations personnel to secure and isolate the unit after a seismic event. The plant document will also require the BWST level be raised to its high limit prior to putting the RO unit into service.

Similarly, we have looked at the possibility of a boron dilution scenario. Boric acid concentration will be logged twice shiftly when the RO process is underway. Administrative and stop work limits will be set in the controlling procedure to preclude approaching Tech Spec limits on boric acid concentration.

Therefore, this change does not create any malfunctions with different results than any previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?

☐ Yes
☒ No

BASIS:

This procedure will control the installation of equipment used to remove suspended and dissolved silica from water in the BWST. This change passively protects fuel in the ANO-1 reactor. The cleaning process and physical alterations will not impact any of the methods of evaluation described in the FSAR.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☒ No

BASIS:

This temporary design change will not modify any plant design bases or methodologies used in the development of the ANO-1 safety analysis. This change relates only to a passive cleaning methodology used to improve BWST chemistry.

If any of the above questions is checked “YES,” obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2004-021

I. OVERVIEW / SIGNATURES

Facility: ANO-Common

Document Reviewed: ER-ANO-2002-1078-006 Original Steam Generator Storage Facility **Change/Rev.: 0**

System Designator(s)/Description: ______

Description of Proposed Change:

This modification involves the addition of a non-safety related reinforced concrete building for the storage of the two original once thru steam generators (OOTSGs), two original RCS hot leg elbows (ORHLEs), and the original reactor vessel closure head (ORVCH) which will be removed from the Unit 1 Reactor Building during the Fall 2005 outage. The Original Steam Generator Storage Facility (OSGSF) will be constructed of cast-in-place and pre-cast reinforced concrete and will have a floor area of approximately 5200 ft^2. It will have two bays, each of which will hold one (1) OOTSG, and a separate compartment to store the ORVCH, original service structure (SS), and control rod drive mechanisms (CRDMs). The ORHLEs will be stored in the OOTSG bay. The OSGSF may also temporarily store the replacement steam generators. The OSGSF will be a new permanent plant structure designed for storage of the above contaminated items. The OSGSF will not interface with any plant structures, systems, or components. To prevent any radioactive releases during the storage period, the OOTSGs and ORHLEs will be drained and all openings in the OOTSGs will be sealed by modification ER-ANO-2002-1078-015, OTSG Removal and ROTSG Preparation/Installation, prior to their removal from the Reactor Building. The ends of the ORHLEs will be sealed by this ER prior to removal from the Reactor Building. The reactor vessel head will be decontaminated, encapsulated, and sealed as necessary by modification ER-ANO-2002- 1078-017, RVCH/SS Shielding, to minimize radioactive releases.

The OSGSF will be located north of the Unit 2 Reactor Building approximately 10 feet to the east of the ANO Unit 2 OSGSF. The OSGSF for ANO-1 will be similar to the existing Unit 2 OSGSF (cast-in-place and pre-cast reinforced concrete). Like the ANO-2 OSGs, the ANO-1 OOTSGs, ORHLEs, and ORVCH are considered contaminated equipment, not radioactive waste, and the OSGSF is designed accordingly. The following discussion provides a basis for this approach.

Background

The Low-Level Waste Policy Amendments Act (LLWPA) of 1985 required that each state provide, either on its own or in cooperation with other states, for the disposal of low-level waste (LLW) generated within the state by December 31, 1992. The LLWPA established an interim access period from January 1, 1986 to January 1, 1993, during which time states and compacts would be allowed continued access to LLW disposal facilities at Barnwell, South Carolina; Hanford, Washington; and Beatty, Nevada. In accordance with the LLWPA, after January 1, 1993, states must be able to store, manage, or dispose of all LLW.

On January 1, 1993, the Beatty land disposal facility closed. Also on January 1 1993, the Hanford facility closed to all states but the Northwest and Rocky Mountain Compact states. The South Carolina Legislature had voted to keep the Barnwell facility open until June 30, 1994 for states that do not belong to the Southeast Compact and until January 1, 1996 for Southeast Compact states. However, on July 1, 1995 South Carolina left the Southeast Compact and opened Barnwell to waste generators in all states except North Carolina. As a result, waste generators in 31 states are no longer forced to store their waste onsite as they have been since July 1, 1994.

When it became apparent that most waste generators would be storing their LLW onsite after January 1993, the NRC Commission directed the NRC Staff to begin a rulemaking which would establish a regulatory framework containing the procedures and criteria that would apply to onsite storage of LLW beyond January 1, 1996. The NRC's intent was to support the goals that had been established by the LLWPA; however, this proposed rule was subsequently withdrawn by the NRC.

On February 3, 1993, the NRC issued a proposed change to the Federal Regulations (proposed rule) which would amend 10CFR Parts 30, 40, 50, and 70 regarding onsite storage of low-level radioactive waste beyond January 1, 1996. The proposed rule would have established procedures and criteria, for onsite storage of LLW that would apply to all categories of LLW generators. Onsite storage of LLW would not be permitted after January 1, 1996 (other than reasonable short-term storage necessary for decay, or for collection or consolidation for shipment off-site, in the case where the licensee has access to an operating LLW disposal facility), unless the licensee could document that it had exhausted other reasonable waste management options.

The proposed regulations would have required that the licensee attempt to contract, either directly or through the state in which the licensee's facility is located, for the disposal of the waste. The proposed regulations would make these requirements standard license conditions for reactor, materials, fuel cycle, and independent spent fuel storage licenses. Licensees would not be required to make a formal submittal to the NRC to show compliance with the requirements of the regulation and make the documentation available to the NRC for inspection.

The proposed rule was not definitive on what constituted LLW. Therefore, it is not clear from the proposed rule whether or not it would apply to large pieces of equipment such as the original steam generators. Documentation is available, however, which would suggest that large contaminated equipment would not be subject to the proposed rule.

The contention that the original steam generators (OSGs) are not considered LLW, but rather contaminated pieces of equipment was suggested by the NRC Staff in SECY-81-383, a Policy Issue Paper, dated June 19, 1981. In late 1980, the NRC drafted a letter to licensees regarding the storage of low-level radioactive wastes at power reactor sites, based on a submittal from the Tennessee Valley Authority to build a life-of-plant, onsite storage facility at Browns Ferry. As a result of the TVA submittal, the NRC Staff proposed a three-tier approach for the licensing of additional storage of low-level reactor wastes generated at reactor sites. The three tiers are: 1) short-term onsite contingency storage capacity which is an additional storage capability provided through modifications and additions that are closely related to existing handling and storage provisions for reactor operations; 2) intermediate onsite contingency storage facilities which are generally separate facilities that are proposed by a utility to provide several years of LLW storage capacity; and 3) life-of-plant onsite storage facilities which are major, separate facilities as exemplified by the Browns Ferry submittal. A package (SECY-80- 511) containing the Staff's proposal, background on the Browns Ferry submittal, the proposed letter to licensees, and a memo on LLW storage at power reactor sites was forwarded to the Commissioners for approval.

Following the issuance of SECY-80-511, the NRC Staff briefed the Commission on the contents of the SECY paper. A number of questions were raised by the Commissioners during the briefing. In the course of developing answers and comments in response to the Commissioners, other issues arose which prompted a revision of the Staff's proposed letter to the licensees informing them of the Staff's plans. These matters were addressed in SECY-81-383. One of the questions asked by a Commissioner and the subsequent Staff response has been extracted from SECY-81-383 and is provided below:

> Question: "What is the effect of this proposal on TMI-2 wastes of low-level classification? Does the EPICOR-2 resins fall in this category, and if so how are they to be treated under this proposal? It looks to me as though this proposal leaves Met Ed with a built-in violation, and precious little way to get out of it. Would it be reasonable to characterize this proposal as applying to LLW from normal operations and to exclude accident-recovery wastes?"
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> Response: "We have not considered this proposed licensing position to be applicable to the TMI-2 situation. It is our intent that the proposal apply to LLW generated from normal operations and to exclude accident-recovery wastes. Another circumstance that would be excluded is the storage of a steam generator that has been removed from service (e.g., Surry) or the storage of other large, contaminated pieces of equipment. We believe that this point can be clarified by modifying the proposal letter to the utilities to indicate that the policy applies to the LLW generated by normal reactor operation and maintenance that conventionally has been shipped to commercial LLW disposal sites."

The proposed letter to the licensees eventually became Generic Letter 81-38, Storage of Low-Level Radioactive Wastes at Power Reactor Sites," which has been referenced in the proposed rule to 10CFR Parts 30, 40, 50, 70, and 72. The NRC Staff did make the following clarification as stated in SECY-81-383: "…for low-level waste generated by normal reactor operation and maintenance at power reactor site." However, the NRC did not provide the specific example that steam generators were excluded as was indicated in the response to a Commissioner's question in SECY-81-383. The Generic Letter states that, for proposed increases in storage capability for LLW generated by normal reactor operation and maintenance at power reactor sites, the safety of the proposal must be evaluated by the licensee under the provisions on 10CFR50.59. The licensee may provide the added capacity, document the 50.59 evaluation, report it to the Commission annually (or as specified in the license), and the five-year license can be renewed, if: (1) the existing license conditions or technical specifications do not prohibit increased storage, (2) no unreviewed safety question exists, and (3) the proposed increased storage capability does not exceed the generated waste projected for five years.

A clarification on the NRC Staff's position as delineated in Generic Letter 81-38 was provided in an NRC Memorandum from L. J. Cunningham, Chief, Radiation Protection Branch, Office of Nuclear Reactor Regulation and P. Lohaus, Chief, Low-Level Waste Management Branch, Office of Nuclear Material Safety and Safeguards to the Directors of the Regions, dated January 31, 1991. Again, the Staff stated that "…radioactive components, such as replaced steam generators or heat exchangers, generated through non-routine maintenance, were not intended to be included within the scope of Generic Letter 81-38."

For previous steam generator replacement projects, the 5-year storage limit defined in Generic Letter 81-38 has not been directly applied to the OSGSF. The reason is, based on previous NRC documentation; the OSGs have not been considered radioactive waste but rather as stored, contaminated equipment to be retained onsite until the plant is decommissioned. This approach has been used by all of the plants that have opted for long-term storage of the OSGs onsite and has been accepted, at least implicitly, by the NRC.

During the proposed rule comment period, D.C. Cook 2, Indian Point 3, Point Beach, and Palisades were contacted for their opinion of the proposed rule and how the rule might impact the future storage of the steam generators that are currently being stored onsite (typically the owner's controlled area). Since they were contacted shortly after the proposed rule was issued, most of the utilities had only begun to assess the potential impact of the proposed rule. However, the utilities did not believe the proposed rule applied to the stored steam generators because they did not consider the steam generators to be LLW, but rather contaminated pieces of equipment. On April 22, 1994 the NRC withdrew the proposed rule which would have amended 10CFR Parts 30, 40, 50, 70, and 72.

Every domestic plant that has replaced its steam generators, with the exception of Millstone 2 and Salem, has stored the OSGs onsite in a non-safety related storage facility. The intention for onsite storage has been clearly noted in various steam generator repair reports (SGRR). Surry 1 & 2, Turkey Point 3 & 4, H.B. Robinson 2, Point Beach 1, and D.C. Cook 2 stated in their SGRR that the steam generators would be stored onsite until the steam generators could be shipped off-site to a burial facility or until the plant was decommissioned. In the cases of Surry, Point Beach, and H.B. Robinson, they stated that the steam generators would remain onsite until the plant was decommissioned. In each case, the NRC reiterated in its SER that the OSGs would be stored onsite and finally concluded that the SGRR was acceptable. Palisades, Indian Point 2 and 3, North Anna 1 & 2, Summer, McGuire 1 & 2, Catawba 1, Byron, Braidwood, ANO-2, and Calvert Cliffs 1 & 2 also are storing the OSGs in an onsite storage facility. These plants did not submit an SGRR to the NRC for approval; however, the onsite storage facility was evaluated under a 10CFR50.59 evaluation.

Conclusion

Based on the following, the OOTSGs, ORHLEs, and ORVCH are considered stored components and not LLW:

- Response to the Commissioner's comment in Secy-80-511.
- NRC memorandum from L. Cunningham/P. Lohaus to directors of the Regions.
- Typical practice of storing original steam generators in OSGSFs for previous steam generator
- replacements.

The NRC provides general guidance on storage facility design in Inspection Procedure 50001, Steam Generator Replacement Inspection (9/6/00) by citing Generic Letter 81-38, Storage of Low Level Radioactive Wastes at Power Reactor Sites. This guidance notes that facility design and operation should assure that radiological consequences of design basis events (fire, tornado, seismic event, flood) should not exceed a small fraction of 10CFR100.

To provide a suitable storage structure for the OOTSGs, ORHLEs, ORVCH, SS, and CRDMs and to provide acceptable radiation shielding, the OSGSF will have the following attributes:

- A nominal three (3) feet thick mat foundation and two (2) feet thick walls and roofs. The roofs will be coated with pitched bituminous membranes that have ten (10) year warranties.
- Sumps located within the OSGSF such that they are accessible from outside the OSGSF. The sumps will primarily collect condensation which may form on the OOTSGs, ORHLEs, and ORVCH.
- Water stops at critical construction joints which will prevent the release or infiltration of water from or into the OSGSF, respectively.

- Two pass doors that will provide personnel access to the OSGSF. Lockable steel wire gates will control access to the doors. Flood barriers will be placed in front of the pass doors to prevent the infiltration of water into the OSGSF through the pass doors.
- A coated floor that will prevent leaching of contamination into the floor.
- Piers located in each OOTSG storage bay across which the saddles supporting the OOTSGs will span. The ORVCH will be supported by two strips of concrete with top of concrete elevations roughly six (6) inches above the floor of the ORVCH compartment. The piers and concrete strips are designed to facilitate off-loading of the OOTSGs and ORVCH. Refer to ER-ANO-2002-1078-009 *ANO-1 SG/RVCH Replacement - Heavy Components Offload, Transport, and Haul Route* for offloading details. The ORHLEs will be supported by the floor slab on specially constructed support stands designed to minimize the radiation dose.
- An access area consisting of compacted soil covered with crushed stone which will accommodate the transporters that will be used to off-load the OOTSGs, ORHLEs, and ORVCH.
- An ORVCH storage compartment that will accommodate the full height of the ORVCH, SS, and CRDMs.
- Concrete keyways and reinforcing bar-splicers located on certain sides of the OSGSF walls and roofs that will facilitate future expansion of the OSGSF.

Rainwater runoff will be directed away from the OSGSF and towards existing drainage systems by grading the area around the OSGSF. Due to the proximity of the Unit 1 OSGSF to the Unit 2 OSGSF, a ditch will be created between the two facilities to direct water away from both facilities. Erosion control measures including the use of temporary and permanent seeding, bailed straw and hay erosion checks, and rip-rap as needed. The grading of the area around the OSGSF and the erosion control measures that will be used will ensure that the OSGSF and its construction will not have any adverse effects on the one-hundred (100) year ditch located to the north of the Unit 2 OSGSF.

Prior to storage in the OSGSF, the OOTSGs will be drained and the open nozzles on the generators will be closed with welded cover plates and shield plugs by separate modifications listed below. After the OOTSGs are drained, there may be a minimal amount of liquid left in them. Since the penetrations of the OOTSGs will be sealed, it is postulated that there will be no release of any residual water remaining in the generators. The ORVCH will also be drained of contaminated liquid prior to storage. A shield plate will be bolted to the bottom of the ORVCH and a gasket will be placed in between the shield plate and the ORVCH. Other openings of the ORVCH and SS will be sealed by various methods such as the use of seal welded cover plates and caulking. For radiation shielding purposes, lead plates will be attached to the ORVCH at various locations. When placed on their stands, the lower portion of the ORHLEs will be sealed by the support stand. The other ends of the elbows will be covered and sealed with cover plates. In addition to being drained, the OOTSGs, ORHLEs, and ORVCH will be encapsulated to seal any remaining loose contamination after decontamination. No airborne release of contamination from the OSGSF is therefore expected. Additional details concerning the preparation of the OOTSGs, the ORHLEs, and the ORVCH for removal and storage are provided in the following modification packages:

- ER-ANO-2002-1078-015 *ANO-1 SG/RVCH Replacement - OTSG Removal and ROTSG Preparation/Installation*
- ER-ANO-2002-1078-017 *ANO-1 SG/RVCH Replacement - RVCH/SS Shielding*

Once the OOTSGs, ORHLEs, and ORVCH are placed in the OSGSF, the open sides of the OSGSF will be closed with pre-cast concrete panels. The panels will be sealed to minimize airflow through the panel joints. No welding will be required to attach the panels to the OSGSF as they will fit into keys built into the ends of the walls of the OSGSF.

Although the OSGSF is sized to accommodate only the OOTSGs, ORVCH, SS, and CRDMs, it could possibly be used to temporarily store the replacement steam generators (RSGs). Before using the OSGSF as a temporary storage area for the RSGs, however, the OSGSF design would be reviewed against the requirements of the RSG storage area as outlined in Section 3.1.1.2 of ANO Specification ANO-M-560 *ANO-1 Steam Generator and Reactor Vessel Closure Head Replacement* and ER-ANO-2002-1078-006 would be revised accordingly.

For security purposes, lighting will be provided in the area in between the Unit 1 and Unit 2 OSGSFs. Although the lighting will be provided for security purposes, it will not be considered security lighting as required in the protected area.

The OSGSF will have no permanent utilities. The OSGSF will be independent (i.e., not physically attached) of other plant structures, systems, and components (SSCs).

A dose assessment analysis will be performed around the OSGSF after installation of the OOTSGs, ORHLEs, and the ORVCH and closure of the OSGSF to verify that the design requirements have been met. Radiation Protection (RP) will review and revise radiological safety plans to include surveillance of the OSGSF to the schedule for non-routine surveillance of radiological areas outside the protected area.

The OSGSF is classified as Non-Safety Related (N) since it does not perform a safety related function and its failure would not prevent the accomplishment of a safety related function. Consistent with Section 5.1.2.2 of the Unit 1 FSAR, the OSGSF is a Seismic Class 2 structure.

The OSGSF is structurally analyzed and designed in accordance with the Uniform Building Code (UBC) 1997. In addition, wind design considering the basic wind velocity for the ANO site as indicated in ASCE 7-02 is considered in the structural analysis and design of the OSGSF to meet NEIL requirements. Refer to ANO Calculation ANO-ER-04-008 *ANO-1 SG/RVCH Replacement - OSGSF Buildin*g. UBC is the building code referenced for use at the ANO site by the Unit 1 FSAR. UBC provides all requirements for concrete and steel design and analysis, testing, and inspection either explicitly or by referencing other codes and documents. The OSGSF is designed such that the dose rates outside of the OSGSF are within the limits of 10CFR20 and 40CFR190 with respect to the requirements of ANO Specification ANO-M-560 *ANO-1 Steam Generator and Reactor Vessel Closure Head Replacemen*t. According to Specification ANO-M-560, the OSGSF must be designed to allow unrestricted access by the general public to the exterior of the building. Section 20.1301 of 10CFR20 states that, “[t]he total effective dose equivalent to individual members of the public ... [shall] not exceed 0.1 rem in a year ...” and that “[t]he dose in any unrestricted area from external sources ... [shall] not exceed 0.002 rem in any one hour.” With respect to occupational doses, the limiting dose requirement of 10CFR20 is provided in Section 20.1208 which states that, “the dose equivalent to the embryo/fetus during the entire pregnancy, due to the occupational exposure of a declared pregnant woman, [shall] not exceed 0.5 rem.”Section 190.10 of 40CFR190 states that for normal operations, “the annual dose equivalent [shall] not exceed 25 [mrem] to the whole body, 75 [mrem] to the thyroid, and 25 [mrem] to any other organ of any member of the public.”

Based on the dose limits of 10CFR20 provided above, the maximum allowable dose, considering a 40 hour/week, 50 week/year (2,000 hour/year) occupancy, at any point on the perimeter of the OSGSF, not including areas on top of the roofs, is 0.05 mrem/hour. As the OSGSF roofs will not be accessed by members of the public, the 10CFR20 dose limit of 100 mrem/year is not applicable for the OSGSF roofs. The areas on top of the OSGSF roofs are therefore assigned a dose limit of 0.1 mrem/hour. If it is assumed that ninety-six (96) percent of the 10CFR20 occupational dose limit for a declared pregnant woman is the dose from other sources at the occupied building closest to the OSGSF, then, considering a 40 hour/week, 50 week/year (2,000 hour/year) occupancy and a pregnancy period of nine (9) months, the 10CFR20 dose limit for dose contributed by the OSGSF at the occupied building closest to the OSGSF is roughly 0.01 mrem/hour. Similarly, if it is considered that 24 mrem/year is the dose from other sources at the point on the exclusion area boundary (EAB) closest to the OSGSF, then, considering a 24 hour/day, 365 day/year (8,760 hour/year) occupancy, the 40CFR190 dose limit for dose contributed by the OSGSF at the point on the EAB closest to the OSGSF is 1 mrem/year.

The dose at various positions within and outside of the OSGSF is provided in Table 5.5-2 and Sections 5.6, 5.7, and 5.8 of ANO Calculation ANO-ER-04-006 *ANO-1 SG/RVCH Replacement - OSGSF Shielding Analysi*s. Table 5.5-2 and Sections 5.6 and 5.7 of Calculation ANO-ER-04-006 indicate that the maximum doses at any point on the perimeter of the OSGSF, the occupied building closest to the OSGSF, and the point on the EAB closest to the OSGSF are within the dose limits for each area as stated above. Because the Unit 1 OSGSF will be adjacent to the Unit 2 OSGSF, the effects of radiation from both facilities are considered in Section 5.8 of Calculation ANO-ER-04-006 which takes into account the maximum dose at the Unit 2 OSGSF walls as provided in ANO Calculation 980642D201-02 *OSGSF Dose Assessment* (Unit 2). Section 5.8 of Calculation ANO-ER-04-006 indicates that the maximum dose that a person would receive standing in the area in between the two facilities is below the 10CFR20 dose limit of 0.05 mrem/hour.

Unrestricted access to the area outside of the OSGSF requires that the area be classified as a Zone 1 radiation area, which, according to Table 11-14 of the Unit 1 FSAR, is an uncontrolled, unlimited occupancy area. As the OSGSF roofs will not be accessed by members of the public, the areas on top of the OSGSF roofs will be classified as Zone 2 radiation areas. Table 11-14 of the Unit 1 FSAR describes a Zone 2 radiation area as a controlled, limited occupancy area. Based on the predicted dose rates (provided in Calculation ANO-ER-04-006) inside the OSGSF east vestibule, the OSGSF west vestibule, and the OSGSF interior, those areas will be designated as radiation Zones 3, 1, and 4, respectively.

The radiological consequences of both the Unit 1 OSGSF and the Unit 2 OSGSF collapsing simultaneously are considered. Although the OSGSF is not a low level radioactive waste (LLRW) facility, the NRC provides general guidance on storage facility design in Inspection Procedure 50001 *Steam Generator Replacement Inspection* by referencing Generic Letter (GL) 81-38, *Storage of LLRW at Power Reactor Site*s. GL 81-38 indicates that facility design and operation should assure that radiological consequences of design basis events (fire, tornado, seismic event, flood) do not exceed a small fraction (ten percent) of the 10CFR100 dose limits. In addition, NRC Regulatory Guide 1.195, which provides methods for evaluating the radiological consequences of design basis accidents, references the dose guidelines of GDC-19 of Appendix A of 10CFR50 for establishing control room dose acceptance criteria. Moreover, the radiological consequences associated with the simultaneous collapse of the Unit 1 OSGSF and the Unit 2 OSGSF are compared to the consequences of postulated accidents for gaseous releases. A breach in the Unit 1 OOTSGs, the Unit 2 Original Steam Generators (OSGs), the ANO-1 ORHLEs, and the Unit 1 ORVCH are considered to be most closely related to the rupture of a tank containing radioactive material. The Waste Gas Tank (WGT) rupture is the limiting event currently evaluated in the Unit 1 SAR. According to Section 14.2.2.7 of the Unit 1 SAR, the quantity of radioactivity contained in a single WGT is limited to a curie value which would prevent a member of the public at the EAB from receiving a total body exposure exceeding 0.5 rem in a two (2) hour period in the event of a WGT rupture.

The radiological consequences of a breach in the Unit 1 OOTSGs and ORVCH and the Unit 2 OSGs are evaluated in ANO Calculations ANO-ER-04-007 *ANO-1 SG/RVCH Replacement - X/Q OTSG & RVCH Drop Analysis* and 980642D203-01 *Doses From Original Steam Generator Drop and OSGSF Failure* (Unit 2), respectively. It should be noted that the ORHLEs were not included in Calculations ANO-ER-04-007. A review of Section 5.0 *Results* and Section VI *Results & Conclusions* of Calculations ANO-ER-04-007 and 980642D203-01, respectively, indicates that the sum of the dose caused by a breach in the Unit 1 OOTSGs and ORVCH and the dose caused by a breach in the Unit 2 OSGs at each location on the ANO site evaluated by both calculations is a small fraction of either the ten (10) percent 10CFR100 or the 10CFR50 dose limit for accidental releases or the maximum predicted dose that a member of the public at the EAB could receive as the result of a WGT rupture. The dose from the ORHLEs is expected to be a small fraction of the dose from the OOTSG and therefore would not be a significant contributor in an OSGSF failure. It is therefore postulated that a design basis event (fire, tornado, seismic event, flood) which causes the simultaneous collapse of the Unit 1 OSGSF and the Unit 2 OSGSF and the subsequent release of contamination from each OSGSF will not result in doses that exceed the ten (10) percent 10CFR100 or the 10CFR50 dose limits for accidental releases or the maximum predicted dose that a member of the public at the EAB could receive as the result of a WGT rupture.

Since significant quantities of water are not expected to be contained in the Unit 1 OOTSGs, ORHLEs, and ORVCH and the Unit 2 OSGs (refer to ANO DCP 980642D201), in the unlikely event of the Unit 1 OSGSF and the Unit 2 OSGSF simultaneous collapse resulting in a contaminated liquid spill, the spill in each OSGSF would be contained within each building and the sumps of each building or, if the slabs of both structures were to fail, within the soil in close proximity to each OSGSF. Any liquid release contained within this soil would then be removed and disposed of in accordance with ANO station procedures.

The subsurface conditions in the area in which the OSGSF will be located are given in ANO Reports 98-R-2013-04 *Subsurface Investigation and Foundation Report, Arkansas Nuclear One, Unit 2 Steam Generator Replacement* and A1-CS-2004-001 *Subsurface Investigation for the ANO-1 Original OTSG and RVCH Storage Facilit*y. Report 98-R-2013-04 indicates that no sub-surface inadequacies exist in the area in which the OSGSF will be located and that for applied loads equal to or less than the allowable bearing capacity of the soil “settlements of mat foundations will be relatively small and well within accepted tolerance limits.” Similarly, Report A1-CS-2004-001 indicates that if the recommended soil bearing pressure is used in the OSGSF design, only small amounts of settlements can be expected. The OSGSF mat foundation is designed considering the allowable soil bearing capacity and other pertinent soil parameters at the OSGSF location. Both Reports 98-R-2013- 04 and A1-CS-2004-001 recommend the same allowable soil bearing capacity. Also, the other pertinent soil parameters provided by both reports are either identical or very similar. Based on the information provided in Reports 98-R-2013-04 and A1-CS-2004-001, the Unit 1 OSGSF is positioned sufficiently far away from the Unit 2 OSGSF such that the interaction effects of the loads from both storage facilities will not compromise the ability of the soil to support each facility. During construction, the soil in the area around the OSGSF will be excavated, backfilled, and compacted. In addition, adequate drainage will be provided during and after construction of the OSGSF. These measures ensure that the soil under the OSGSF will adequately accommodate the loads imposed on it.

According to Section 2.4.4.3 of the Unit 1 SAR, the maximum predicted flood elevation for the OSGSF is 361 *ft* with the possibility of water being splashed, from wave action, 10 *ft* above a static water level of 358 *ft*. Section 5.1.6 of the Unit 1 SAR indicates that Seismic Class 1 structures are designed for the maximum probable flood elevation of 361 *ft* and that Seismic Class 2 structures are designed for a design flood elevation of 338 *ft*. Nonetheless, although the OSGSF is a Seismic Class 2 structure, to prevent water from entering the OSGSF in the event of a flood, all possible flow paths for water will be sealed. Water stops will be placed in vertical construction joints up to an elevation of 361 *ft* and in horizontal construction joints which are below or at an elevation of 361 *ft*. Removable steel plates which will be sealed will also be placed in front of the OSGSF pass doors up to an elevation of 361 *ft*. In addition, the joints of the OSGSF door panels will be sealed. Splashing of water above 358 *ft* will not affect the OOTSGs, ORHLEs, and the ORVCH as they will be enclosed by the OSGSF walls. With respect to the loads that could be imposed on the OSGSF during flooding, Calculation ANO-ER-04-008 *ANO-1 SG/RVCH Replacement - OSGSF Building* indicates that the OSGSF is capable of withstanding the hydrostatic forces that could be imposed on the OSGSF in the event of a flood.

Although the OSGSF will be sealed to prevent water from entering the OSGSF, the OOTSGs, ORHLEs, and the ORVCH will be positioned inside the OSGSF and sealed to minimize the release of contamination if water does enter the OSGSF in the event of a flood. The OSGSF floor elevation will be 357 *ft* 6 *in*. The bottom of the saddles supporting the OOTSGs will be supported on piers with top of plate elevations of 362 *ft*. The ORVCH will be supported on concrete strips that rise 6 *in* above the floor. If water enters the OSGSF in the event of a flood, the OOTSGs will therefore not be in contact with water and 3 *ft* of the ORVCH could potentially be under water. Release of contamination from the OOTSGs into the water will therefore not occur. Because 3 *ft* of the ORVCH could be under water, the ORVCH will be sealed to minimize release of contamination. In a similar manner as ORVCH, the ORHLEs are stored on racks that support the ORHLEs 5" above the concrete floor. The only opening that water would have access to is the pipe opening that was once connected to the OOTSG hot leg nozzle. This opening will be sealed by the support stand and will be further sealed by RTV or other suitable material.

Although there will be minimum combustibles associated with the OSGSF (e.g., encapsulant and roofing materials), the OSGSF will be an unoccupied facility, without electrical power and without an ignition source. Based on the low combustible loading of the OSGSF and the absence of an ignition source, a fire event is very unlikely.

The OSGSF will be a stand-alone, non-safety related facility that will not interface with permanent plant structures or systems and will not be connected to any existing structures. In the event of a fire in any plant area, the SSCs of the operating facility, and administrative controls relied upon to ensure ANO's ability to achieve and maintain a safe shutdown condition will not be affected by the OSGSF. Although the Unit 1 OSGSF will be adjacent to the Unit 2 OSGSF, as the Unit 2 OSGSF has minimal combustible materials and no ignition source or electric power (refer to ANO DCP 980642D201), the Unit 2 OSGSF will pose a minimal fire related threat to the Unit 1 OSGSF and vice versa.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: FFN 04-021 Rev 1)**	**Sections I, II, and IV required**

Preparer: Wayne R. Wasser / **ORIGINAL SIGNED BY WAYNE WASSER** / Adecco Tech / SG/RVCH / 9-12-05
Name (print) / Signature / Company / Department / Date

Reviewer: Doyle G. Adams / **ORIGINAL SIGNED BY DOYLE ADAMS** / EOI / SG/RVCH / 09-12-05
Name (print) / Signature / Company / Department / Date

OSRC: J. R. Eichenberger / **ORIGINAL SIGNED BY J. R. EICHENBERGER** / 9-15-05
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	Unit 1 FSAR - 5.1.2.2.1, 5.3.5.7, 11.1.3.3.8, 11.2.3, 11.2.4.7, 11.2.4.8 (new) 11.4and Figure 11-10 (new). Table of Contents' Pages: xxii, xl, 5-iv, 11-ii, and 11-vi Unit 2 FSAR – 11.5.6, 12.1.2.12, 12.1.3.4, 12.5; Figures 2.5-17, 12.1-6A, and 12.1-13
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES", perform an Exemption Review per Section III OR perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☐ **Yes**
☒ **No**

If "yes," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. Basis

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.3.2 for guidance.

Operating License/Technical Specifications

The ANO-1 and ANO-2 Operating Licenses, Technical Specifications, Technical Specification Bases, and Technical Requirements Manuals were reviewed to determine the impact from the proposed addition of the ANO-1 OSGSF. The design and implementation activities of ER-ANO-2002-1078-006 to construct an onsite facility for the storage of the ANO-1 OOTSGs, ORHLEs, and ORVCH fully comply with the ANO-1 and -2 Operating Licenses, Technical Specifications, NRC Orders, and Technical Requirements Manual (TRM). The Tech Specs and TRM provide operational limitations on systems directly related to the safe operation of ANO-1 and -2. None of the Tech Specs or TRM requirements address site structures such as the existing storage facility for the ANO-2 OSGs. The planned installation of the OSGSF for storage of the ANO-1 OOTSGs, ORHLEs, and ORVCH will have no impact on any plant structures, systems, or components nor will it invalidate Tech Spec or TRM requirements. Therefore, no changes to the ANO -1 or -2 Tech Specs, Tech Spec Bases, or TRM are required.

FSAR

The ANO-1 and ANO-2 FSARs are impacted by the addition of the ANO-1 OSGSF:

Unit 1 FSAR

A brief discussion of the Unit 1 and 2 Original Steam Generator Storage Facilities (OSGSFs) will be added to Sections 5.3.5.7 and 11.1.3.3.8. Section 5.1.2.2.1 will be revised to reference Section 5.3.5.7 in addition to referencing the Unit 2 FSAR. The information in Section 11.2.4.7 "Other Plant Areas" will be transferred to a new section, Section 11.2.4.8. Section 11.2.4.7 will be changed to discuss OSGSF shielding. A brief discussion of the computer code used in the OSGSF shielding calculation (refer to ANO Calculation ANO-ER-04-006) will be added to Section 11.4. A new figure, Figure 11-10, will be added to show the radiation zones in and around the Unit 1 OSGSF. A discussion concerning Figure 11-10 will be added to Section 11.2.3.

Unit 2 FSAR

Discussion of the Unit 2 OSGSF in Sections 11.5.6 and 12.1.3.4 will be revised to indicate that there are two OSGSFs, one for each unit. The discussion of OSGSF shielding in Section 12.1.2.12 will be revised to indicate that there is more than one OSGSF. The computer code used in the OSGSF shielding calculation will be added as a reference in Section 12.5. The Unit 1 OSGSF will be added to Figures 2.5-17 and 12.1-13 as shown on DWG. No. C-2002 (DRN 04-00925). The title of Figure 12.1-6a will be revised to indicate that the radiation zones shown in the figure are those for the Unit 2 OSGSF.

The construction and use of the OSGSF will have no impact on any plant SSCs and will not result in SSCs being utilized or controlled in a manner that is outside the reference bounds of the design basis as described in the FSAR or is inconsistent with the analyses or descriptions in the FSAR. Therefore, per the guidance provided in NEI 96-07, Rev. 1, the construction and use of the OSGSF would not be considered a test or experiment.

The OSGSF will be constructed in an area north of the Unit 2 Reactor Building. Neither the construction nor use of this facility will have an impact on activities related to the Independent Spent Fuel Storage Installation.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
Autonomy – LRS 50.59 – Common	"steam generator storage facility", "OSGSF", SG NEAR10 storage, OTSG NEAR10 storage, "steam generator*" NEAR10 storage, "site plan*", "abbreviatio*", "general arrangement", "shield*", "radioactive equip*", "offsite dose*", "off-site dose*", "low level" NEAR20 "radioactiv*", "low level" NEAR20 waste, offsite NEAR20 storage, "off-site" NEAR20 storage, "flood*", "tornado missile*", "81-38", "50001", "10cfr20", "10 cfr 20", "40cfr190", "40 cfr 190", "title 10" NEAR5 "part 20", "title 40" NEAR5 "part 190", "10 cfr" NEAR5 "part 20", "40 cfr" NEAR5 "part 140", "hot leg", "hot legs", hotleg*, elbow*

LBDs/Documents reviewed manually:

ANO-1 FSAR:

Sections 1.2 ,1.4.2, 1.4.4, 1.4.5, 2.4.4, 5.1.2.2, 5.3.5, 11.1, 11.2, 11.4, Figures 1-2 through 1-11

ANO-2 FSAR

3.3, 3.4, 11.5, 12.1, 12.5; Table 1.7-1; Figures 2.5-17, 12.1-6A, 12.1-13, Figures 1.2-1 through 1.2-11

5. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☐ **Yes**

☒ **No**

If "YES", list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1]See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: _____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10 CFR 50.59 Review Program Guidelines for guidance.

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   ER-ANO-2002-1078-006, Original Steam Generator Storage Facility provides for the addition of a reinforced concrete building that will be used for the interim storage of the ANO-1 OOTSGs, ORHLEs, and the ORVCH. This facility will be located adjacent to the ANO-2 OSGSF. The facility will not interface with any plant structures, systems, or components (SSCs) and will therefore not affect the performance or reliability of ANO-1 or ANO-2 SSCs. Construction of the OSGSF may be performed during any Unit 1 or Unit 2 mode of operation without creating any adverse impact to SSCs. The OSGSF will not serve to mitigate any accidents currently evaluated in the ANO-1 or ANO-2 FSARs. The OSGSF does not have the apability of initiating any accidents currently evaluated in the ANO-1 or ANO-2 FSARs.

   The OSGSF is designed to conform to the UBC for non-safety related structures. The radiological consequences of a simultaneous collapse of the Unit 1 OSGSF and the Unit 2 OSGSF during a design basis event (fire, tornado, seismic, flood) were considered and it was demonstrated that the offsite dose would be bounded by a Waste Gas Tank rupture (Unit 1 FSAR Section 14.2.2.7). Installation and use of the Unit 1 OSGSF will not affect the frequency of occurrence of natural phenomena such as tornados, floods, earthquakes, or high winds.

   Therefore, the installation and use of the OSGSF will not increase the frequency of occurrence of an accident previously evaluated in the FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The proposed permanent modification to install a facility to store the Unit 1 OOTSGs, ORHLEs, and ORVCH will have no impact on the operation of either Unit. Since the OSGSF will be remote from Units 1 and 2 and will not be connected to any structures, systems or components, construction and operation of the OSGSF will not adversely impact the operation of any structures, systems, or components that are important to safety.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The FSAR accident analysis does not address events related to the storage of contaminated equipment onsite. The proposed permanent modification to add an OSGSF to store the OOTSGs and ORVCH will have no impact on the radiological consequences of any accidents previously evaluated in the FSAR. In the event of an OSGSF collapse, the radiological consequences of a breach in the OOTSGs and ORVCH are evaluated in ANO Calculation ANO-ER-04-007 *ANO-1 SG/RVCH Replacement - X/Q OTSG & RVCH Drop Analysis*. The calculation indicates that the dose resulting from a breach in the OSGs and ORVCH is within regulatory limits. It should be noted that the ORHLEs were not included in Calculation ANO-ER-04-007. However, the dose from the ORHLEs is expected to be a small fraction of the dose from the OOTSG and therefore would not be a significant contributor in an OSGSF collapse.

According to Section 2.4.4.3 of the Unit 1 SAR, the maximum predicted flood elevation for the OSGSF is 361 *ft* with the possibility of water being splashed, from wave action, 10 *ft* above a static water level of 358 *ft*. Section 5.1.6 of the Unit 1 SAR indicates that Seismic Class 1 structures are designed for the maximum probable flood elevation of 361 *ft* and that Seismic Class 2 structures are designed for a design flood elevation of 338 *ft*. Nonetheless, although the OSGSF is a Seismic Class 2 structure, to prevent water from entering the OSGSF in the event of a flood, all possible flow paths for water will be sealed. Water stops will be placed in vertical construction joints up to an elevation of 361 *ft* and in horizontal construction joints which are below or at an elevation of 361 *ft*. Removable steel plates which will be sealed will also be placed in front of the OSGSF pass doors up to an elevation of 361 *ft*. In addition, the joints of the OSGSF door panels will be sealed. Splashing of water above 358 *ft* will not affect the OOTSGs, ORHLEs, and the ORVCH as they will be enclosed by the OSGSF walls. With respect to the loads that could be imposed on the OSGSF during flooding, Calculation ANO-ER-04-008 *ANO-1 SG/RVCH Replacement - OSGSF Building* indicates that the OSGSF is capable of withstanding the hydrostatic forces that could be imposed on the OSGSF in the event of a flood.

Although the OSGSF will be sealed to prevent water from entering the OSGSF, the OOTSGs, ORHLEs, and the ORVCH will be positioned inside the OSGSF and sealed to minimize the release of contamination if water does enter the OSGSF in the event of a flood. The OSGSF floor elevation will be 357 *ft* 6 *in*. The bottom of the saddles supporting the OOTSGs will be supported on piers with top of plate elevations of 362 *ft*. The ORVCH will be supported on concrete strips that rise 6 *in* above the floor. If water enters the OSGSF in the event of a flood, the OOTSGs will therefore not be in contact with water and 3 *ft* of the RVCH could potentially be under water. Release of contamination from the OOTSGs into the water will therefore not occur. Because approximately 3 *ft* of the ORVCH and ORHLEs could be under water, the RVCH and ORHLEs will be sealed to minimize release of contamination.

Therefore, addition of the OSGSF will not have an impact on the radiological consequences of any accidents previously analyzed in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The permanent addition of the OSGSF to the plant site and storage of the OOTSGs will not change, degrade, or prevent actions described or assumed in any malfunction of equipment important to safety.

The permanent addition of the OSGSF to the plant site and storage of the OOTSGs will not change, degrade, or prevent actions described or assumed in any previously evaluated accident analysis. Construction of the OSGSF will have no adverse effect on plant flood levels. Due to their massive weight, the OOTSGs, ORHLEs, and ORVCH will not become tornado missiles. The SSCs and administrative controls relied upon in the event of a fire in any plant area to ensure ANO's ability to achieve and maintain a safe shutdown condition will not be affected. Given the negligible amount of combustibles that will be inside the facility and the lack of an ignition source, a fire in the OSGSF is very unlikely.

Therefore, the installation and use of the OSGSF will not increase the consequences of a malfunction of SSCs important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

In the unlikely event of the collapse of both OSGSFs, a breach in the Unit 1 OOTSGs and ORVCH and the Unit 2 OSGs are evaluated in ANO Calculations ANO-ER-04-007 *ANO-1 SG/RVCH Replacement -X/Q OTSG & RVCH Drop Analysis* and 980642D203-01 *Doses From Original Steam Generator Drop and OSGSF Failure* (Unit 2), respectively. The offsite dose consequences associated with the collapse of the OSGSF are compared to the consequences of postulated accidents for a gaseous release. (It should be noted that the ORHLEs were not included in Calculation ANO-ER-04-007. However, the dose from the ORHLEs is expected to be a small fraction of the dose from the OOTSG and therefore would not be a significant contributor in an OSGSF collapse.) For assessing offsite dose consequences, an OSG/ORVCH release is considered most closely related to the rupture of a tank containing radioactive material. The waste gas tank (WGT) rupture as described in the Unit 1 FSAR Section 14.2.2.7 is the limiting event currently evaluated in the FSAR for accidental gaseous releases. The radiological consequences of a failure of both of the Unit 1 OOTSGs and Unit 2 OSGs and the Unit 1 ORVCH and ORHLEs are a small fraction of the 10CFR100 guideline values for accidental releases and are less than the consequences of the WGT as described in FSAR Section 14.2.2.7 (WB exposure <0.5 Rem in a 2 hour period for a member of the public at the Exclusion Area Boundary). The dose consequences of a collapse of the OSGSFs have been demonstrated to be within the applicable regulatory guidelines and less than the comparable licensing basis accident currently evaluated in the FSAR.

Since significant quantities of water are not expected to be contained in the Unit 1 OOTSGs, ORHLEs, and ORVCH and the Unit 2 OSGs (refer to ANO DCP 980642D201), in the unlikely event of the Unit 1 OSGSF and the Unit 2 OSGSF simultaneous collapse resulting in a contaminated liquid spill, the spill in each OSGSF would be contained within each building and the sumps of each building or, if the slabs of both structures were to fail, within the soil in close proximity to each OSGSF. Any liquid release contained within this soil would then be removed and disposed of in accordance with ANO station procedures.

Therefore, the installation and use of the OSGSF will not create the possibility of an accident of a different type than any previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The proposed permanent modification to install a facility to store the Unit 1 OOTSGs, ORHLEs, and ORVCH will have no impact on the operation of either unit. Since the OSGSF will be remote from Units 1 and 2 and will not be connected to any structures, systems or components, construction and operation of the OSGSF will not adversely impact the operation of any structures, systems, or components that are important to safety.

Therefore, construction and operation of the OSGSF will not create the possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

The proposed permanent modification to install a facility to store the Unit 1 OOTSGs, ORHLEs, and ORVCH will have no impact on the operation of either unit. Since the OSGSF will be remote from Units 1 and 2 and will not be connected to any structures, systems or components, construction and operation of the OSGSF will not adversely impact any of the fission product barriers.

During the time that the OOTSGs, ORHLEs, and ORVCH are a part of the reactor coolant system, they are considered to be part of the reactor coolant pressure boundary and therefore, fission product barriers. After removal from the reactor coolant system, the OOTSGs, ORHLEs, and ORVCH will be drained of liquids and sealed and will no longer be considered pressure retaining components.

Therefore, construction and operation of the OSGSF will not adversely impact the pressure retaining capability of the RB or other fission product barriers.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

The computer code MCNP4C was used for the shielding calculations. This code is a generalized geometry, time dependent, Monte Carlo transport code that can be used to model complex radiological shielding models. In NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems,” the NRC notes that MCNP is an appropriate computer program to be used for shielding analysis and is widely known and recognized for performing such analyses. Since the OSGSF is an on-site contaminated component storage structure similar to an ISFSI, the NRC concurrence is considered applicable.

The radiological consequences of a breach in the OOTSGs and ORVCH is evaluated in ANO Calculation ANO-ER-04-007 *ANO-1 SG/RVCH Replacement - X/Q OTSG & RVCH Drop Analysi*s. The ELISA-2 computer code used to calculate offsite doses is universally approved by the NRC for applications of the type used here. The ELISA-2 computer code uses dose conversion factors from Federal Guidance Reports 11 and 12. According to U.S. NRC Regulatory Issue Summary 2001-19, “Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests,” the NRC staff considers use of these dose conversion factors to be an acceptable change in methodology that does not warrant prior review.

Therefore, the use of the MCNP4C and ELISA-2 computer codes does not represent a departure from a method of evaluation previously evaluated in the FSAR.

If any of the above questions is checked “YES”, obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure ENS-LI-113.

ANO 50.59 Evaluation Number

2004-025

I. OVERVIEW / SIGNATURES

Facility: ANO-1

Document Reviewed: Procedure 1409.638, Unit 1 IS Natural Convection Cooling Test **Change/Rev.: 0**

System Designator(s)/Description: **SW, Service Water; FP, Fire Protection**

Description of Proposed Change:

Natural convection cooling is the safety related means of removing waste heat from operation of service water pumps P-4A, B and C and associated components in the Unit 1 Intake Structure in the event the normal forced ventilation exhaust fans VEF-25 and VEF-32 are lost due to failure or loss of offsite power. The Unit 1 SAR credits "architectural openings" to provide a pathway for this natural convection flow. Louvered doors on the ground level and elevated open roof plugs on the roof above each SW pump motor are currently credited as the architectural openings in the free convection design.

During wintertime, the louvers in doors 171 and 172 (HVD-226 and HVD-227) are closed to prevent freezing of components on the ground level. These actions result in degraded free convection capability, which could potentially result in degrading the SW pumps should an event occur where normal cooling capability is assumed to be lost (i.e. Appendix R fire event or a DBA). This condition is documented in CR-ANO-C-2004-1848. Current system operability is maintained by keeping door louvers in the open position, but this operating position could eventually cause freezing problems this winter.

Transient heatup analysis using the GOTHIC computer code indicates that if site ambient temperatures are 70 F or below and door louvers are closed with two of three SW pumps in operation, Operations has at least 3-4 hours to access and open the door louvers to establish free convection cooling before a limiting temperature of 140 F is exceeded in the Intake Structure.

This Activity is a test to validate the calculation results and determine if other architectural openings in the Intake Structure such as the idle pump roof plug opening, the normal roof intake opening or connecting ductwork can provide alternate means of cooling the SW pump motors. In addition, since the diesel driven fire pump in an adjacent room will energize on a loss of offsite power, this pump will be run as well to determine thermal effects on the intake structure and on SW pump cooling. The expected test duration is approximately 12 hours. Since this test places the Intake Structure in a condition not explicitly described in the SAR, it is considered a Test or Experiment. A 50.59 Evaluation is performed and attached.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: FFN 04-025)**	**Sections I, II, and IV required**

Preparer: David MacPhee / **ORIGINAL SIGNED BY DAVID MACPHEE** / EOI / DE / 11-09-04
Name (print) / Signature / Company / Department / Date

Reviewer: Alexander McGregor / **ORIGINAL SIGNED BY ALEXANDER MACGREGOR** / EOI / DE / 11-09-04
Name (print) / Signature / Company / Department / Date

OSRC: J. R. Eichenberger / **ORIGINAL SIGNED BY J. R. EICHENBERGER** / 11-11-04
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See LI-101 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☐	☒	
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III OR perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the LBD change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113.			

[1] If "YES," see LI-101. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☒ **Yes**
☐ **No**

If "yes," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

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3. Basis

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.3.2 for guidance.

<u>Operating License/Technical Specifications</u> This Activity is well below the level of detail contained in these documents. Administrative controls within this Activity are provided to ensure Technical Specification requirements for the SW systems are not violated.

<u>FSAR</u> This test activity is not described in the FSAR. The SAR does note that "architectural openings" in the Intake Structure are credited for free convection, but does not define these openings in detail. Therefore, this portion of the SAR is unaffected by this Activity. Administrative controls in this activity are provided to ensure FSAR Chapter 9, Appendix 9D requirements for Fire Protection systems are not violated.

<u>Tests or Experiments considerations</u>. This activity is considered a Test since the natural ventilation features will be purposely altered in a way that will place the intake structure natural ventilation in a condition contrary to the SAR intent. A 50.59 Evaluation is attached.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
Autonomy Index: 50.59-Unit-1	"intake structure", "service water pump*", "fire water pump*", "jockey pump*", "ventilation test", "heatup test", "temperature test", "startup test", "experiment*", "P-4A", "P-4B", "P-4C", "P-6A", "P-6B", "P-11".

LBDs/Documents reviewed manually:

ANO-1 FSAR: Chapter 9, sections 9.3, 9.7, 9.8, Appendix 9D; Chapter 9 Figures and Tables, Chapter 13.

5. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☐ **Yes**
☒ **No**

If "YES", list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: _____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10 CFR 50.59 Review Program Guidelines for guidance.

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.

☐ Yes
☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?

   ☐ Yes
   ☒ No

   BASIS:

   Analyzed accidents in FSAR Chapter 14 were reviewed. The ANO-1 SAR does not contain a loss of service water as an accident. This activity affects the ambient temperature in the Intake Structure and can impact components in the Intake Structure. No components in the Intake Structure are analyzed accident initiators. No other potential accident initiators in the plant are impacted by this change.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

   ☐ Yes
   ☒ No

   BASIS:

   The ambient temperatures in the intake structure will be allowed to rise beyond levels maintained by the normal ventilation systems, but administrative controls will be in place to assure that the maximum temperatures will remain below previously qualified limits. Safety related components in the Intake Structure are qualified to at least 140 F per Calculation 93-R-1006-01. Temperatures in the SW pump motor area will be limited to 130 F maximum. If this temperature is exceeded, the door louvers will be opened as outside conditions permit, free convection pathways will be re-established, and normal ventilation will be placed in operation. Service water pumps and associated electrical devices will be operated in accordance with approved procedures, with the exception of actions normally taken per Annunciator Corrective Action (ACA) procedure 1203.012I for SW pump and SW winding high temperature alarms. Since the most probable cause of these alarms is lack of proper ventilation, the corrective action for these alarms only is specified in the workplan. The specified action is to open the door louvers and restore forced ventilation, restoring the test conditions to the pre-test configuration. Other actions in the existing ACA that may be necessary to clear these pump high temperature alarms are beyond the scope of this workplan. Other SW pump alarm conditions and ACA response will not be affected by this workplan.

   An increase in ambient temperature above qualified limits results in some loss of oil viscosity in the motor bearing oil resulting in reduced bearing life, and loss of life in motor winding insulation. These degradation mechanisms do not result in outright failure of SW water pump motors except under extreme temperature conditions for an extended time. Other safety related electrical components are qualified for higher ambient temperatures and in similar fashion higher temperatures above qualified limits can result in loss of life. Since the test will control pump room temperatures below qualified limits, the probability of SW pump motor failure or other safety related electrical failure due to temperature effects is considered extremely unlikely.

   The fire water pumps and associated components are designed for a maximum normal operating temperature of 110 F per Bechtel purchase specification M-32. Diesel fire pump operation in room ambient temperatures higher than 110 F has been evaluated in Calculation 93-D-5015-05. At temperatures up to at least 120 F, no adverse effects are expected. For the purpose of this test, if 110 F is reached in either fire pump room the operating fire pump will be secured in accordance with established procedure. If temperatures in the fire pump rooms do not subsequently decrease, the test will be terminated and the door louvers will be opened and normal ventilation established. In no event will qualified temperatures be exceeded for safety related equipment, and operation above 110 F for fire rated components in the intake structure will be limited to short durations. Since ambient temperatures will be monitored under strict administrative controls below the qualified limits of this equipment, there will not be an increase in the likelihood of occurrence of a malfunction of affected components.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The mitigative function of the service water pumps and associated components will not be adversely affected by this test. The room temperatures will not be allowed to exceed the design limits of the SW pumps and associated electrical circuits. SW pumps and associated components will be operated in accordance with approved procedures (except as noted above in Question 2 response) during the test and the idle SW pump will be maintained as an operable standby. Since the ability of the SW system to mitigate accidents is not adversely affected by this test, there is no increase in the possible dose consequences resulting from postulated accidents.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

Since ambient temperatures will be maintained within qualified limits of the service water pumps and associated components, no new failure modes or malfunctions not previously considered will be introduced by this testing. The consequences of an analyzed malfunction of a service water pump or associated component in the intake structure remain unchanged.

In similar fashion, the diesel driven fire pump will be operated in accordance with established procedure and will be secured and placed in normal automatic operation mode if established ambient temperature limits are exceeded. Thus proper operation of fire water pumps and associated components is not adversely affected by this Activity. Fire water pumps are used to mitigate fire events in support of safe shutdown.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

Adequate administrative controls will be in place for this activity such that systems used to mitigate accident or fire events remain properly functional. The SW pump room temperatures will be monitored, therefore, the test will not create a loss of service water or related accident. No new failure modes are introduced and no other systems important to safety are impacted by this Activity.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The only design parameter affected by this activity is the intake structure ambient temperature. SW pump operation is not otherwise affected. Very high ambient temperatures above qualified limits for an extended time may result in pump motor failure due to winding degradation or motor bearing failure due to degraded oil lubricants. The results of this ambient temperature induced motor failure will not be changed by this Activity. All other SW pump design requirements will be protected to avoid potential pump failures. Since temperatures will be maintained within the qualified limits of the equipment and this is the only design parameter affected, postulated failure modes for safety related equipment in the Intake Structure remain unchanged. In the event the higher temperature does affect a pump motor or associated electrical switchgear or electrical component, the failure mode is pump motor failure or electrical component failure, which are analyzed events. Thus, postulated failures will not cause a different result.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?

☐ Yes
☒ No

BASIS:

This Activity does not affect fuel cladding or RCS boundary parameters or limits, and does not affect the containment design pressure. Systems used to protect these limits will continue to be properly supported by operable service water and fire water systems.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☒ No

BASIS:

This activity has no impact on any method of evaluation as described in the FSAR as determined by screening.

If any of the above questions is checked "YES", obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure ENS-LI-113.

ANO 50.59 Evaluation Number

2004-029

I. OVERVIEW / SIGNATURES

Facility: **ANO - Unit 1**

Document Reviewed: ER-ANO-2002-1078-014, ANO-1 SG/RVCH Replacement - 2" & Under Piping and Supports **Change/Rev.:** 0

System Designator(s)/Description:

Description of Proposed Change:

To support the ANO-1 Replacement Project, ER-ANO-2002-1078-014, ANO-1 SG/RVCH Replacement - 2" & Under Piping and Supports, provides for the removal and reinstallation of small diameter secondary system piping (with diameters of 2" or less) and associated supports from each once through steam generator (OTSG), and for the permanent removal of a primary drain line segment of piping and associated supports that have been made obsolete as a result of the improved OTSG design (as discussed in Section 2.2.1.2 of ANO Calculation 021381E101-59, Rev. 0, "Enhanced Once Through Steam Generator (EOTSG) Report for EOI ANO Unit 1" (AREVA/FANP Document No. 77-5018868-00), which is included in ER-ANO-2002-1381-000, ANO-1 SG/RVCH Replacement – ROTSG Design/Qualification). Insulation for various piping/lines will be removed and replaced or discarded as necessary.

The OTSG Secondary Drain lines function to drain steam generator (SG) levels during start-up, adjust levels during wet lay-up, perform chemistry adjustments during shut down, and blowdown the OTSG during start-up after cold shutdown. The OTSG Vent line's primary function is to allow for venting of the SG secondary side during OTSG fill and drain operations. The OTSG primary drain line functions to drain primary water from the OTSG that has accumulated in the OTSG below the primary outlet nozzles.

Secondary Drain and Vent Lines (EBD-19, EBB-5 and EBB-6)

Each OTSG is provided with eight (8) drain lines which provide for secondary draining from the tubesheet and the steam annulus areas, and secondary vent lines connected to the OTSG at one nozzle that exits through the side of the upper tubesheet.

The secondary system connections include:

- Two (2) 1 ½" drain nozzles at elevation 381'-1 1/8"
- Two (2) 1" drain nozzles at elevation 352'-4" (includes attached sample lines)
- Four (4) 1 ½" lower tube sheet (LTS) drain nozzles at elevation 347-10" (includes attached sample lines) It also supports instrumentation tubing to be removed/reinstalled under ER-ANO-2002-1078-020, ANO-1 SG/RVCH Replacement – Tubing/Supports and Instrumentation.
- One (1) 1 ½" secondary vent line nozzle at elevation 402'-2 ¾"

In order to remove sections of the OTSGs Secondary Drain lines, the piping will be severed at each nozzle approximately 3" from the SG shell. The piping will be cut off downstream of the nozzle in a manner that provides sufficient clearance to permit the removal of the OOTSGs and the installation of the ROTSGs. The open nozzle ends on the OOTSG nozzles will be capped with seal welded plugs. The installation of the plugs is addressed in ER-ANO-2002-1078-015, ANO-1 SG/RVCH Replacement – OTSG Removal and ROTSG Preparation/Installation.

Before reinstalling the Secondary Drains Lines, the remnant of the OOTSGs nozzles will be removed and the end of the nozzles will be discarded. The remaining piping will be reinstalled after the ROTSGs are installed.

The ROTSG LTS drain nozzles are supplied with 2" nozzles in lieu of the 1 ½" nozzles currently installed on the OOTSG to allow for increased drain capacity capability should larger drain piping be installed in the future. The 2" nozzles are provided with a reducing insert for reattachment to the existing 1 ½" piping. The 2" x 1 ½" reducing inserts are provided with the ROTSG and are qualified in ER-ANO-2002-1381-000.

The four (4) LTS drain nozzles on each OOTSG were originally supplied with plain end connections. Presently, the OOTSGs have socket weld pipe fittings welded to the plain end of each nozzle. The four (4) nozzles, on the ROTSGs, will come with socket weld reducers welded to the nozzles. A new section of pipe will be welded to each of the ROTSGs socket weld reducers and the other end of the new pipe will be welded to the portion of drain piping that was temporarily removed prior to the removal of the OOTSG.

A segment of sample lines attached to drain lines will be cut and temporarily removed along with the drain lines to allow for removal of the OOTSGs and installation of the ROTSGs. After the ROTSGs are installed the removed sample lines will be reinstalled. Existing insulation on the two sample lines will be removed and new calcium silicate insulation will be reinstalled for personnel protection.

Sections of instrument tubing are supported from LTS drain lines. The instrument tubing supports are to remain attached to the drain lines when they are temporarily removed to allow for OTSG removal. Removal and reinstallation of the instrument tubing is addressed by ER-ANO-2002-1078-020.

The original vent piping will be cut approximately 3” from the SG as well as further down the pipe. This cut portion of vent piping will be temporarily removed so that the OOTSG can be removed and the ROTSG can be installed. A section of each pipe from the nozzle end will be removed and discarded. The remainder of the temporarily removed piping will be reinstalled after the ROTSG is installed. The open nozzle ends on the OOTSG nozzles will be capped with seal welded plugs as described in ER-ANO-2002-1078-015.

Each vent line will be disconnected from its support. These supports will remain attached to the Upper Lateral Restraint (ULR) steel. Removal of the ULR steel will be addressed in ER-ANO-2002-1078-015. The piping cuts are made near an existing support or will be temporarily supported in the interim until reattachment.

The vent nozzles on both OOTSGs are plain pipe end at a 45° angle from horizontal. The vent nozzles on both ROTSGs are socket welded connections at a 32° angle from horizontal. The discarded section of pipe will be replaced with a bent pipe for reconnection of the piping to the new nozzle orientation.

The new ROTSG socket welded vent nozzles are approximately 5” longer than the existing nozzles.

On the secondary vent piping, the existing calcium silicate insulation within the scope of this ER will be removed and replaced with new calcium silicate after the piping is reinstalled. The insulation to be removed will be disposed of.

Primary Drain Line (CCA-13)

The OTSG Primary Drain piping to be permanently removed consists of one (1) 1” drain line that exits the bottom of the OOTSG lower head, which connects to the Primary Drain nozzle that is being eliminated on the ROTSG design per ANO Calculation 021381E101-59, “Enhanced Once Through Steam Generator (EOTSG) Report for EOI ANO Unit 1” (AREVA/FANP document No. 77-5018868-00). The ROTSG has a flat bottom bowl head that allows for complete draining through the primary outlet nozzles.

In order to remove the OTSG Primary Drain line, the piping shall be severed approximately 3” below the OTSG nozzle and approximately 7” to 10” from the existing 2” x 2” x 1” tee that is connected to the 2” Primary Drain header for each line. The open nozzle ends on the OOTSG nozzles will be capped with seal welded plugs as described in ER-ANO-2002-1078-015. The removed Primary Drain piping and insulation will be discarded as directed by Radiation Protection. The end of piping CCA-13-1” left open by the downstream cut will be permanently capped. A pipe cap will be welded to the open end of the pipe.

Two valves (RBD-6A and RBD-6B) and supports (CCA-13-H5, CCA-13-H9, and CCA-13-H10) will be permanently removed along with the drain line. This change causes Licensing Basis Document impacts to FSAR Figures 4-1, 7-2, and 11-1.

Code Reconciliation

The code reconciliation document, “Code Reconciliation – 2” and Under Piping and Supports”, in ER-ANO-2002-1078-014, evaluates and justifies the use of alternative Construction Code requirements for repair/replacement activities to be performed during the Arkansas Nuclear One Unit 1 Steam Generator Replacement (SGR) Outage. The repair/replacement activities will be performed in accordance with ASME Section XI, 1992 Edition, no Addenda.

Seismic II/I

The interim piping configuration for the affected piping (due to temporary removal of portions of the piping) has been reviewed as to its impact on any safety-related/Seismic Category I equipment (i.e. Decay Heat Removal System, EFIC, Refueling Canal Level Indication) in the unlikely event of a seismic occurrence during Mode 5 (Cold Shutdown with the OTSGs no longer required for RCS decay heat removal), and Mode 6 (Refueling) when the affected piping will be severed and in a temporarily supported condition. EFIC tubing and components located below and in the vicinity of the affected piping are required to be operable during Modes 1, 2, & 3; therefore, work on the affected piping during Modes 5 & 6 will not result in any operability concerns with respect to Seismic II/I interaction.

The Primary Drain piping will be removed when the reactor is defueled and no safety-related/Seismic Category I equipment in the vicinity of the Primary Drain piping is required to be operable; therefore, there is no impact on any safety-related/Seismic Category I equipment or piping from a Seismic II/I standpoint.

The Secondary Vent (effected piping is 1 1/2"Φ Sch. 80 and less) and the Secondary Drain/Sample (effected piping is 2"Φ Sch. 80 and less) will be removed while Unit 1 is in Mode 5 (Cold Shutdown) once ANO Operations has determined that the OTSGs are no longer required for RCS decay heat removal, or in Mode 6 (Refueling). In the severed and temporarily supported condition, the Secondary Vent and Secondary Drain/Sample piping will not interact with any (required to be operable) safety-related/Seismic Category I SSCs during any postulated seismic event due to the location of the affected piping and the Modes in which the subject work will be performed.

Plant Modes/ Implementation Restrictions

No implementation of work activities affecting the OTSG secondary side pressure boundary for this ER is allowed while the plant is in Modes 1, 2, 3, 4, or 5 (Power Operation/Startup/Hot Standby/Hot Shutdown/Cold Shutdown) with the OTSG required for decay heat removal.

As determined by construction, activities such as pre-fabrication, preparation, and pre-staging of equipment and materials outside the reactor building may be performed at any time before or during the OTSG replacement outage.

Removal and reinstallation of the insulation on the OTSG Upper Tubesheet Vent, Primary Drain, and Sample lines shall be performed during Mode 5, 6, or with the reactor defueled.

The removal and reinstallation of the Secondary Drain/Sample and Vent lines will be performed while Unit 1 is in Mode 5 (Cold Shutdown) once ANO Operations has determined that the OTSGs are no longer required for RCS decay heat removal, Mode 6 (Refueling), or with the reactor defueled.

The secondary side of the OTSG, including drain and vent piping will be depressurized and drained as required prior to piping severance.

The removal of the OTSG Primary Drain lines and capping the remaining line is restricted to Unit 1 reactor defueled and the RCS drained and declared out of service. The Primary Drain line modifications shall be completed prior to entry into Mode 6 (Refueling).

All work within the scope of this ER should be completed prior to entering Mode 5 (with the SGs required for decay heat removal).

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 04-029)**	**Sections I, II, and IV required**

Preparer: John Pearman / **ORIGINAL SIGNED BY JOHN PEARMAN** / NAGI / SG-RVCH / 12-05-04
Name (print) / Signature / Company / Department / Date

Reviewer: Doug Barborek / **ORIGINAL SIGNED BY DOUG BARBOREK** / EOI / SG-RVCH / 12-05-04
Name (print) / Signature / Company / Department / Date

OSRC: J.R. Eichenberger / **ORIGINAL SIGNED BY J.R. EICHENBERGER** / 1-24-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	ANO-1 FSAR Figures 4-1, 7-20 and 11-1. An LBD change document has been generated.
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR? ☐ **Yes** ☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change <u>AND</u> initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. <u>Basis</u>

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

<u>Operating License/Technical Specifications (TSs)</u>

The activity involves removal and reinstallation of small piping and associated supports connected to the OTSG to their design configuration, and the deletion of the steam generator lower head primary drain line piping and associated supports. The drain line piping will not be needed since the replacement OTSG doesn't have a lower head drain nozzle to connect to. The lower head drain will be unnecessary since the ROTSG has a flat, self-draining bowl.

The removal and reinstallation of the piping while Unit 1 is in Mode 5 (Cold Shutdown) once ANO Operations has determined that the OTSGs are no longer required for RCS decay heat removal, Mode 6 (Refueling), or with the reactor defueled does not impact any of the required Modes of Operability of the OTSGs or other safety-related SSCs. The Primary Drain Line Piping is not discussed in the Operating License, Technical Specifications, Tech Spec Bases, Technical Requirements Manual, or Core Operating Limits Report; therefore, these documents are unaffected.

<u>FSAR</u>

The Primary Drain Line piping to be eliminated is shown on ANO Drawing M-230 Sh 1, "Piping & Instrument Diagram – Reactor Coolant System", which is included in the ANO-1 FSAR as Figures 4-1 and 7-20. Drawing M-230 Sh 1 will be revised to show this change via DRN 04-00389 to ANO Drawing M-230 Sh 1, "Piping & Instrument Diagram Reactor Coolant System", Revision 106. ANO-1 FSAR Figure 11-1 is also impacted by the elimination of the primary drain line piping. Drawing M-214 Sh 3 will be revised to show this change via DRN 04-01857 to ANO Drawing M-214 Sh 3, "Piping & Instrument Diagram-Clean Liquid Radioactive Waste", Revision 17. FSAR Section 4.2.1.1 discusses the Reactor Coolant System (RCS). It refers to Figure 4-1 that shows the RCS schematic and the two steam generator drains/drain lines. (The same figure is also shown as Figure 7-20.) The FSAR does not discuss the use of the steam generator lower head drain line. Therefore, the proposed deletions and changes to the steam generator lower head drain piping will only require a change to the FSAR figures noted and will not adversely affect a design function described in the FSAR, or impact the facility or any procedure described in the FSAR.

The removal of the lower head SG drain nozzle (not drain line) is the result of a design enhancement to the Replacement (Enhanced) OTSG. ANO Calculation 021381E101-59, "Enhanced Once Through Steam Generator (EOTSG) Report for EOI ANO Unit 1" (AREVA/FANP document No. 77-5018868-00), describes the EOTSG design and fabrication, and describes the evaluation of the use of these SGs at ANO Unit 1. A 10 CFR 50.59 evaluation (Section 4.0 of AREVA/FANP document No. 77-5018868-00), was prepared to support the conclusion that the EOTSGs do not involve the criteria listed in Section c.2 of 10 CFR 50.59. The 10 CFR 50.59 evaluation addressed the component hardware design differences between the OTSGs and the EOTSGs, and confirmed that the EOTSG design is in accordance with appropriate design technical requirements and that the EOTSGs do not involve the criteria listed in Section c.2 of 10 CFR 50.59. The 10 CFR 50.59 will include revisions to ANO affected documents including changes to FSAR Table 4-4 (SG Design Data) and Paragraph 4.2.2.2 revised to reflect the new SG design to include changes to SG drain information and elimination of SG lower head drain.

ER-ANO-2002-1381-000, ANO-1 SG/RVCH Replacement - ROTSG Design/Qualification, documents the design and qualification of the AREVA EOTSGs and affected Nuclear Steam Supply System (NSSS). The purpose of ER-ANO-2002-1381-000 is to incorporate SG documents into the EOI Document Control System.

Test or Experiments Consideration

The activity involves removal of small piping and associated supports connected to the OTSG and their reinstallation to their design configuration. It also will permanently remove the OTSG Primary Drain piping that is not needed due to the elimination of the OTSG lower head drain in the enhanced design of the OTSGs. The function the lower head drain piping served will be accomplished by use of the self-draining bowl of the new SG. These activities do not require components, systems, or groups of systems to be operated in modes for which they were not previously analyzed. The temporary removal of the OTSG small bore piping and the elimination of the Primary Drain piping is not considered a test or experiment.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LRS 50.59 – Unit 1	“drainline”, “generator w/10 drain*”, “OTSG w/10 drain*”, “RCS w/10 drain*”, “drain/fill”, “drain/flush”, “drain/letdown”, “drain w/10 nozzle”, “bottom w/10 head”, “secondary drain”, “secondary vent”, “primary drain”, “small pipe”, “generator w/10 piping”, “OTSG w/20 piping”, “CCA-13*”, “RBD-6*”, “lower tube sheet drain”
LBDs/Documents reviewed manually:	
ANO Unit 1 SAR	Sections 4.2.1.1, 4.2.2.2, 7.1.4, 10.4.9, 11.1, 16.1.3, 16.2.20 Chapters 1, 4 Figures 4-1, 4-5, 4-13, 4-15, 7-20, 11-1 Tables 4-4
ANO Unit 1 Technical Specifications	Sections 3.3.11, 3.4, 3.9

5. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☒ **Yes**
☐ **No**

If “YES,” list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

ER-ANO-2002-1078-014 impacts FSAR Figures due to the deletion of a segment of RCS Drain piping as implemented by this ER. This segment of drain piping cannot be deleted unless the OTSG primary drain nozzle has been eliminated by ER-ANO-2002-1381-000 and related licensing basis document changes submitted for ANO-1 approval for implementation. A Post Action within the Engineering Response Database (ERD) will be generated to verify prior to the scheduled outage start date that ER-ANO-2002-1381-000 has been approved for implementation.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☒	☐	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☒ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: _____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10 CFR 50.59 Review Program Guidelines for guidance.

50.59 Evaluation performed as requested by Licensing (see Section IV). Section III (50.59 Evaluation Exemption) not required.

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.	☐ Yes ☒ No

Does the proposed Change:

1.	Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?	☐ Yes ☒ No

BASIS:

Most accidents evaluated in ANO-1 FSAR Chapter 14 occur while the unit is at power. The Fuel Handling Accident occurs in the spent fuel pool in the Auxiliary Building or in the Reactor Building, while the unit is not at power during Mode 6 – Refueling. The activities in this ER occur in the Reactor Building during shutdown, and disconnect and reconnect piping and their associated supports to their design configuration, in addition to permanently removing a segment of primary drain piping made obsolete by the enhanced OTSG design. The piping is small diameter (2 in. diameter and under) drain and vent piping connected to the OTSG. It is to be removed from the OTSG after the OTSG is declared out of service and no longer required for decay heat removal or RCS pressure boundary. The removal and reinstallation of the OTSG small diameter piping and associated supports during unit shutdown as described in the ER, would not affect any system in a way that could lead to a fuel handling accident.

When the unit returns to operation, the piping will be in its design configuration. The configuration has changed for the primary drain piping due to removal of a drain line segment of piping. The function of the permanently removed primary drain piping is provided by the self-draining OTSG bowl of the enhanced OTSG design. The reconnected piping will perform in the same manner as before, does not affect overall system performance, and will not initiate or increase the possibility of occurrence of any FSAR evaluated accident described when the unit is at power.

Therefore, it is concluded that the changes in ER-ANO-2002-1078-014 would not increase the frequency of an accident previously evaluated in the FSAR, during implementation of or as a result of the ER.

2.	Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?	☐ Yes ☒ No

BASIS:

The piping and supports affected by ER-ANO-2002-1078-014 are associated with the OTSG. The OTSG and associated piping are not in service and are not required to be in service at the time of piping removal/reconnection as described in this ER. Therefore a malfunction of the OTSG during implementation of ER-ANO-2002-1078-014 should not be considered.

The removed secondary piping and supports will be reinstalled to their design configuration. Also, the obsolete primary drain line segment of piping previously connected to the primary drain at the bottom of the OTSG will be removed by this ER. (The function of the permanently removed primary drain is provided by the self-draining flat bottom bowl head design of the enhanced OTSG, which will allow for complete draining through the primary outlet nozzles. The enhanced OTSG design, including the elimination of the OTSG primary drain, is described in ANO Calculation 021381E101-59, Rev. 0, "Enhanced Once Through Steam Generator (EOTSG) Report for EOI ANO Unit 1" (AREVA/FANP No. 77-5018868-00).

The activities involved in this ER return the SG small bore piping to their design configuration. The design configuration has changed for the primary drain piping due to removal of a drain line segment of piping. These changes do not change the operation or function of the OTSG, or increase the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The ER disconnects and reconnects OTSG small diameter secondary system piping and supports and removes obsolete OTSG primary drain line piping, during the SGR outage. The small bore piping associated with the OTSG does not serve to mitigate any accidents. The piping and supports will be returned to the design configuration. The obsolete piping removed will be discarded as directed by Radiation Protection. The OTSG and associated piping are not required to be in service at the time of piping removal/reconnection as described in this ER.

The implementation or results of these activities do not alter the consequences (increase radiological dose) of a fuel handling accident (unit off-line) or other accidents (unit at power) described in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The piping and supports affected by this ER are associated with the OTSG. The OTSG and associated piping are not required to be Operable and are not required to be in service at the time of piping removal/reconnection as described in this ER. Therefore a malfunction of the OTSG during implementation of ER-ANO-2002-1078-014 should not be considered. After implementation of the ER, the secondary piping will have been reinstalled to its design configuration and the obsolete primary drain line segment of piping will have been removed, and the function of the OTSG will be unchanged as a result. The RCS will have a system pressure test prior to returning the OTSGs and piping to service.

Therefore, any consequences as a result of a possible malfunction of the OTSG or other structure, system, or component (SSC) important to safety previously evaluated in the FSAR, will not be increased.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

This ER involves the activities of disconnecting and reconnecting piping and supports to replace a piece of equipment. The unit will be shut down and the associated component (the piece of equipment – OTSG) will be out of service and released by Operations. All piping welds performed as part of ER-ANO-2002-1078-014 will be inspected in accordance with ASME Code requirements. There is no accident created to cause failure of the piping beyond what is currently licensed for ANO-1. This activity has no impact on the existing accident analyses in the ANO-1 FSAR, and does not create a possible accident that would likely happen while the unit and OTSG are out of service, or when the piping has been returned to its design configuration and unit operation resumes.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The piping and supports affected by this ER are associated with the OTSG. The OTSG and associated piping are not in service and are not required to be in service at the time of piping removal/reconnection as described in this ER. All piping welds performed as part of ER-ANO-2002-1078-014 will be inspected in accordance with ASME Code requirements and these piping systems will receive post installation testing in accordance with ASME Code requirements. Therefore a malfunction of the OTSG during implementation of ER-ANO-2002-1078-014 should not be considered. After implementation of the ER, the secondary piping will have been returned to their design configuration, and the obsolete primary drain line segment of piping will have been removed. As a result, the function of the OTSG and the remaining RCS drain line piping will be unchanged, and the ER would not have created a possibility for a malfunction of a SSC important to safety.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

ANO-1 fission product barriers include fuel cladding, RCS Boundary, and Containment Pressure.

Fuel Cladding

All open ends of the removed piping/valves shall be covered to prevent foreign materials from entering the system. The removed and remaining portions of the Secondary Drain and Vent piping will be tagged for traceability. Storage of the Secondary Drain and Vent components shall be stored in accordance with design and quality program requirements. Prior to closing the system, the cleanliness of the Secondary Drain and Vent lines shall be verified in accordance with design and quality program requirements.

The obsolete primary drain line segment of piping will be disconnected, removed, and subsequently discarded (as directed by Radiation Protection). The end of piping left open by the downstream cut will be capped. Prior to closing the system, the cleanliness of the Primary Drain lines shall be verified to meet design and quality program requirements. This will prevent any foreign materials from entering the primary system and possibly damaging the fuel cladding.

ER-ANO-2002-1078-014 will not result in a design basis limit for the fuel cladding barrier as described in the FSAR being exceeded or altered.

RCS Boundary

The Secondary Drain and Vent piping is connected to the secondary side of the steam generators and must be capable of maintaining the secondary side pressure boundary. The OTSG Primary Drain is connected to the primary side of the OTSG and must be capable of maintaining the primary side pressure boundary.

The affected sections of the Primary Drain, 1 ½" Secondary Drain and Vent piping will be pressure tested in accordance with ASME Section XI as part of the OTSG primary/secondary pressure test prior to entry into Mode 2 (Startup). Although exempted from ASME Section XI (< 1" for Class 2), the affected sections of the 1"Secondary Drain piping will be pressure tested as part of the ROTSG secondary side pressure test. The OTSG secondary side boundary will be maintained until released by Operations. The RCS primary boundary will be maintained until the OTSG is drained and released by Operations.

ER-ANO-2002-1078-014 will not result in a design basis limit for the RCS Boundary barrier as described in the FSAR being exceeded or altered.

Containment Pressure

Removal and reconnection of the OTSG secondary drain and vent lines will have no impact on the Reactor Building. Likewise, removal of the OTSG primary drain will not impact the Reactor Building.

ER-ANO-2002-1078-014 will not result in a design basis limit for the Containment Pressure barrier as described in the FSAR being exceeded or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

The ER disconnects and reconnects OTSG small diameter secondary system piping and supports and removes obsolete OTSG primary drain line piping previously connected to the primary drain at the bottom of the original OTSG. There is no safety analysis, design analysis or other methods of evaluation being changed as a result of this ER.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-001

I. OVERVIEW / SIGNATURES

Facility: ANO - Unit 1

Document Reviewed: ER-ANO-2003-0063-000 **Change/Rev.:** 0

System Designator(s)/Description: Chemical Addition and Sampling System (Zinc Injection)/RCS

Description of Proposed Change:

The scope of this review includes ER-ANO-2003-0063-000. This proposed change involves the addition of a Zinc Injection System. The system consists of a mixing tank, recirculation pump, two 100% metering pumps, and associated piping, valves, and instruments. The zinc, in the form of zinc acetate, is injected into the Reactor Coolant Make-Up Tank upstream of the Primary System Makeup Water Filters, via the discharge line of the Hydrazine Pump. The addition of zinc to the RCS is performed as a means to reduce radiation fields within the RCS. CALC-A1-ME-2004-005, "Assessment of Zinc Injection for Dose Reduction at ANO Unit 1 provides a complete discussion to the benefits and impacts of zinc injection in the primary system.

Check the applicable review(s): **(Only the sections indicated must be included in the Review.)**

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-001)**	**Sections I, II, and IV required**

Preparer: Lawrence D. Theriault / **ORIGINAL SIGNED BY LAWRENCE THERIAULT** / DP Eng, Ltd / 1-12-05
Name (print) / Signature / Company / Department / Date

Reviewer: Christopher A. Davenport / **ORIGINAL SIGNED BY CHRIS DAVENPORT** / DP Eng, Ltd / 1-12-05
Name (print) / Signature / Company / Department / Date

OSRC: J.R. Eichenberger / **ORIGINAL SIGNED BY J.R. EICHENBERGER** / 1-24-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	FSAR Figure 9-4, DRN 04-1339 (P&ID M-233 Sh. 1); New FSAR Figure; DRN 04-1373 (P&ID M-233 Sh. 2); FSAR Figures 1-2, A-1, DRN 04-1376 (Equip. Loc. Dwg. M-2); FSAR Section 4.2.3.6, FSAR Section 9.2; FSAR Table 9.4; FSAR Table 9-6; FSAR Figure 11-2, DRN 04-1476 (P&ID M-213, Sh. 1).
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☐ **Yes**
☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change <u>AND</u> initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. <u>Basis</u>

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

<u>Operating License/Technical Specifications (TSs)</u>

The Unit 1 SAR was reviewed, and it was found that nowhere is zinc injection into the RCS mentioned. The proposed modification adds a non nuclear-safety related zinc injection system, consisting of a tank, a recirculation pump, two injection pumps, and associated piping, valves and instrumentation, all mounted on a skid. No other part of the LBDs, Technical Specifications or Operating License contains detail on zinc in the RCS. The following FSAR Sections are impacted as a result of the change:

- FSAR Figure 9-4, Chemical Addition and Sampling System P&ID is revised to depict the connection point for the new Zinc Injection system.
- A new FSAR figure (P&ID) is added to depict the new Zinc Injection system.
- FSAR Figures 1-2, A-1, Equipment Location drawings are revised to show the new equipment skid location.
- FSAR Section 4.2.3.6, RCS Chemical Control, is revised to add discussion regarding Zinc Injection.
- FSAR Section 9.2, Chemical Addition and Sampling Systems, is revised to add description of Zinc Injection system.
- FSAR Table 9-4, Reactor Coolant Quality, is revised to add limits of zinc concentration in primary water chemistry.
- FSAR Table 9-6, Chemical Addition and Sampling System Component Data, is revised to add new major components of the Zinc Injection system.
- FSAR Figure 11-2, Dirty Radioactive Waste Drainage and Filtration P&ID is revised to depict the connection points for the drains from the zinc injection skid.

The water chemistry of the RCS is affected by the zinc injection system. The application of zinc causes a restructuring of the system oxides and a resulting release of divalent cationic species, including iron, cobalt, and nickel. Previous plant experience has demonstrated that a large portion of these releases, in particularly cobalt, are in the form of insoluble species. During the course of an operating cycle, these increases in soluble and insoluble cationic species may lead to increased crud deposit on the fuel. In order to monitor the zinc injection program and ensure that excessive amounts of crud are not being transported to the fuel, the practice of monitoring the at-power iron, nickel, and suspended solids concentration in the RCS will be adopted.

No tests or experiments are proposed that will operate the zinc injection system, the RCS, or the plant in a manner, mode or configuration that will invalidate the SAR. The purpose of the zinc injection into the RCS is only to reduce radiation fields within the RCS.

The proposed change does not change any ANO-1 RCS or Chemical Addition and Sampling System functions.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LRS Search: 50.59 – Unit 1	zinc; chemistry; chemical addition; reactor coolant NEAR20 chemistry; hydrogen; oxide; crud; and fuel
LBDs/Documents reviewed manually:	U1 SAR: Section 3A.4.2, "Fuel Rod Analysis", Chapter 4, "Reactor Coolant System", Section 6.6, "Post LOCA Hydrogen Control", Section 9.2, "Chemical Addition and Sampling Systems," Table 9-4, "Reactor Coolant Quality," Table 9-6, "Chemical Addition and Sampling System Component Data", and SAR Figure 9.4.

5. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☐ **Yes**
☒ **No**

If "YES," list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☒	☐	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☒	☐	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☒	☐	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a “yes” box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: _____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10 CFR 50.59 Review Program Guidelines for guidance.

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation <u>ONLY</u>? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.

☐ Yes
☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The addition of zinc into the RCS water chemistry does not introduce the possibility of a change in the frequency of an accident because the addition of zinc into the primary water chemistry is not an accident initiator, and no new failure modes are introduced. The proposed zinc injection system is non-safety related and does not inject directly into the RCS. It will be powered from non-safety related power supplies and will be installed to meet all applicable ANO-1 design basis considerations, such as Seismic category II/I requirements. The zinc injection skid will be located on the west side of the boric acid mix tank platform (floor El. 404'-0", Room 157). There is existing equipment in this area, but the zinc injection skid will not affect the operation of any existing equipment. All new tubing and conduits for the system will be routed and located along the walls, away from existing equipment. The location of the tie-in point for the zinc injection system will be on the discharge line of the hydrazine pump, upstream of valve CA-56. The hydrazine pump only operates during an outage. The zinc injection system only operates when the plant is in operation. Therefore, connecting the zinc injection system to the hydrazine pump discharge will not affect the operation of the hydrazine pump. There are no failure modes associated with the equipment that can initiate an accident. The Moderator Dilution Accident, Loss-of-Coolant Accident, and Steam Generator Tube Failure are the only conditions that may be credibly affected as a result of a zinc injection equipment failure that added excess zinc to the primary water chemistry.

The frequency of a Moderator Dilution Accident is not increased by the installation of the zinc injection system because the zinc solution is mixed into the Reactor Coolant Makeup Tank (RCMT). The final return flow to the RCS is therefore determined by the make-up pump flow, which is independent of the zinc injection system. For this reason, a failure of the zinc injection system would not result in an increased feed to the RCS, only an increase in the RCMT volume. The small increase in volume that would result from adding an entire carboy of zinc (~5) gallons is insignificant compared to the volume of the tank. Therefore, zinc addition will have no affect on this accident condition as it is analyzed in the FSAR.

A Loss-of Coolant Accident, as it is evaluated in the FSAR, involves the double-ended rupture of a 36-inch diameter RCS pipe. Zinc injection has no negative impacts on RCS materials, including piping, and may have a small beneficial effect on Primary Water Stress Corrosion Cracking (PWSCC) initiation. Therefore, zinc injection is not expected to increase the frequency of this accident condition. A Steam Generator Tube Failure, as evaluated in the FSAR, involves a steam generator tube leak when the reactor is operating with 1% failed fuel. Zinc injection has no negative impacts on the RCS materials, including SG tubes. In fact, zinc injection may serve to decrease the likelihood of this type of accident by slowing the PWSCC initiation rate. Therefore, zinc injection is not expected to affect either the frequency or consequences of this type of accident.

The effect of zinc on the wetted surfaces in the RCS and associated equipment does not introduce a mechanism to accelerate degradation of equipment or components. It is generally accepted that zinc is effective at mitigating the extent and consequences of general corrosion of both austenitic stainless steels and nickel-base alloys in primary water. Zinc may offer some additional beneficial effects regarding the frequency of accidents as there is laboratory data and plant experience that zinc slows the initiation of PWSCC of Alloy 600 at higher concentrations. Potential minimization of PWSCC of steam generator tubes requires higher concentrations than currently planned for the zinc injection. Although the concentrations currently proposed are lower than required for PWSCC minimization, this information shows that zinc is not an initiator of this type of failure mode at the lower concentrations currently proposed.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The addition of zinc into the RCS does not introduce the possibility of a change in the likelihood of occurrence of a malfunction of an SSC important to safety because the addition of zinc into the reactor coolant is not an initiator of any new malfunctions and no new failure modes are introduced. The effect of zinc on the wetted materials in the RCS, fuel cladding, vessel internals, steam generator tubes, and associated equipment does not introduce a mechanism to accelerate degradation of equipment or components. There is no indication of accelerated fuel corrosion at the low concentrations planned for use nor is there any industry history of fuel related failures attributed to zinc in the reactor coolant. In addition, a strategy has been developed to minimize any increase in Crud Induced Power Shift (CIPS) risk due to the addition of zinc in ANO-1. This strategy includes delaying zinc addition 3 months to ensure that a protective oxide is formed on new fuel and starting zinc addition at lower concentrations (5 to 10 ppb) than have already been used in plants utilizing zinc injection, which have not experienced significant levels of CIPS. Although, the program maximum planned concentration of 10 ppb may be exceeded during normal startup/shutdown and off normal conditions; zinc concentration spikes of short duration are expected to be within the industry experience base and will not cause immediate or eminent fuel failure. The zinc injection system is non-safety related. A failure of any portion of the system does not directly or indirectly impact any SSC important to safety.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The addition of zinc into the RCS does not introduce the possibility of a change in the consequences of an accident because the addition of zinc to the RCS will not affect the ability of the reactor protection system (primarily reactor trip functions) from operating properly during accident conditions. Additionally, zinc injection will not significantly affect the environmental consequences of the accidents evaluated in the FSAR because those accidents that resulted in an environmental release assumed that the reactor was operating with 1% failed fuel, and in some cases that the fuel rod gap activity was released. In all cases, the additional activity in the RCS due to zinc injection (primarily an increase in soluble and insoluble radiocobalts) was insignificant when compared to that already assumed in the evaluation. In addition, because cobalt is relatively non-volatile, the releases to the environment would be miniscule, as the evaluated releases are gaseous. It can therefore be concluded that zinc injection does not create an increase in the radiological consequences of an accident previously evaluated in the FSAR.

Additionally, fuel failure resulting from degradation of the cladding is not a new failure mode and is bounded by existing fuels analyses. At the planned injection concentration, no degradation of fuel performance is expected, based on Framatome fuel corrosion evaluations and the fact that there have been no PWR zinc-related fuel failures in the industry.

The consequences of a Moderator Dilution Accident described in the FSAR are not increased by the installation of the zinc injection system. Because the zinc acetate solution will be added to the Reactor Coolant Makeup Tank, the actual dilution flow into the RCS is independent of the zinc injection system, and controlled by the primary make-up pumps.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The addition of zinc into the RCS does not introduce a change in the consequences of a malfunction of an SSC important to safety. Zinc, in low concentrations, has proven to be benign, or even slightly beneficial, to RCS materials and systems during numerous US PWR applications. Based on the available test and industry data, there is no reason to believe that the malfunction of an SSC would be exacerbated due to zinc injection. Based on Framatome's fuel evaluation and current industry experience, zinc will not result in fuel failures beyond that already analyzed in the FSAR. Additionally, zinc injection will have no impact on reactor building integrity, since zinc will not be in contact with the isolation points in other than minute, trace concentrations. Therefore, the consequences of a malfunction of an SSC important to safety are not affected by zinc injection.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The addition of zinc to the reactor coolant does not create the possibility for an accident of a different type than any previously evaluated in the FSAR. Based on previous industry experience, the addition of zinc, in larger concentrations, was thought to potentially adversely affect cladding oxidation. However, the results of oxide thickness measurements indicate that zinc does not have a statistically significant effect on cladding corrosion. Based on previous experience, zinc addition had no other adverse effects on the fuel. With the small, controlled addition of zinc, crud buildup and cladding corrosion are expected to be acceptable and insignificant. The expected increase is well within the experience base at other plants with higher coolant temperatures. With a negligible increase in cladding oxidation and fuel rod temperatures being the only effects of the zinc addition, new types of accidents are not introduced.

Additionally, previous investigations conducted by EPRI have shown that low concentrations of zinc in the RCS will have no negative impacts on 304 stainless steel and Alloy 600, and in fact serve to reduce the corrosion rate of these materials. This data is equally applicable to Alloy 690 and other grades of stainless steel. In addition, several US PWRs are currently injecting zinc in both low concentrations (< 10 ppb) and high concentrations (20-40 ppb) with no negative impacts reported in any of the materials of construction. Based on the above information, one can conclude that zinc will have no negative impacts on the corrosion of the general RCS materials of construction and therefore will not create a possibility for an accident of a different type than previously evaluated in the FSAR.

While zinc injection does involve the installation and operation of new equipment, the installation and operation of the equipment will be governed by approved procedures. These procedures will minimize any risks associated with new operations.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The addition of zinc does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR. With the small, controlled addition of zinc, crud buildup and increased cladding oxidation are expected to be acceptable and insignificant. The expected increase is well within the experience base at other plants with higher coolant temperatures. In addition, Post-LOCA Hydrogen Generation condition is not affected by the addition of depleted or natural zinc acetate into the RCS. In order for zinc to take part in hydrogen producing reactions, zinc would have to be reduced to the elemental state. It can be empirically shown that the RCS environment is not sufficiently reducing to produce zinc metal from the added zinc acetate, and without the production of zinc metal, the zinc will not take part in the hydrogen generation during accident conditions. Thus, the addition of zinc acetate will not result in an increase in the generation of hydrogen in Containment as analyzed in the FSAR.

The experience to date with German and U.S. PWRs has shown that Zinc excursions above the target range are expected during shutdowns and possibly during load swings as a result of zinc release from RCS surfaces. These excursions are not considered detrimental when accompanied by reasonable actions to return the zinc residual to within the control range. These actions would include terminating chemical addition.

Additionally, previous investigations conducted by EPRI have shown that low concentrations of zinc in the RCS will have no negative impacts on 304 stainless steel and Alloy 600, and in fact serve to reduce the corrosion rate of these materials. This data is equally applicable to Alloy 690 and other grades of stainless steel. In addition, several US PWRs are currently injecting zinc in both low concentrations (< 10 ppb) and high concentrations (20-40 ppb) with no negative impacts reported on any of the materials of construction. Based on the above information, one can conclude that zinc will have no negative impacts on the corrosion of the general RCS materials of construction and therefore will not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?

☐ Yes
☒ No

BASIS:

From a fuel integrity perspective, with a negligible increase in cladding oxidation and fuel rod temperatures being the only possible effects of the zinc addition, adequate margins will remain to the existing limits. The margin of safety for the fuel is addressed by the following review of the applicable design criteria and the impact of the zinc addition of the fuel evaluation.

Internal Hydriding

The internal hydriding criterion is aimed at the introduction of moisture or other hydrogenous sources inside the fuel rod during fabrication. The hydrogen limit is unaffected by the zinc addition and margin of safety does not change.

Cladding Collapse

Cladding collapse may occur if the initial fuel rod cladding ovality, and/or differential pressure are sufficiently high during operation. In addition, the formation of large gaps in the fuel pellet column due to excessive densification may permit the cladding to collapse since the fuel column normally provides support for the cladding. The potential for a small increase in cladding oxidation or crud buildup will not significantly affect cladding creep, ovality or fuel densification. The mechanism that can cause cladding collapse is therefore not impacted by the zinc addition.

Overheating of Cladding

The design basis for fuel rod cladding overheating is to preclude departure from nucleate boiling (DNB) during normal operation and anticipated operational occurrences (AOO). Compliance with these criteria is confirmed as part of the reload thermal hydraulic analysis. The potentially small increase in oxidation and crud buildup is expected to be within the normal range of operating experience and will have no effect on the occurrence of DNB. Overheating of the cladding is also protected during the LOCA transient by demonstrating compliance with the 2200ºF temperature limit from 10 CFR 50.46. The analyses are performed on a plant-specific basis for each fuel type used in the reactor Framatome ANP uses conservative oxide layers at the beginning of the LOCA transient to exacerbate the metalwater reaction during the accident and increase the calculated peak clad temperature (PCT). Any additional initial oxidation or crud buildup due to the zinc could slightly reduce the oxidation rate and this would have little effect (favorable if any) on the LOCA analyses.

Overheating of Fuel Pellets

Fuel temperature is limited to prevent fuel melting from occurring during normal operation and AOO. The fuel melt limit is conservatively reduced based on test data and as a function of burnup. The fuel melt limit and margin of safety associated with the limit will not be affected by the zinc addition. Current calculations indicate a much larger margin than the potentially small increase in fuel temperatures due to oxidation and crud buildup. The temperature increase associated with an additional 2 microns of zinc oxide on the fuel is expected to result in a temperature increase that is less than 2ºF. The change in temperature due to zinc addition is expected to be negligible in comparison to the current margin such that there is minimal change in margin of safety. If zinc oxide is introduced onto once-burnt or twice-burnt fuel, the effect of the deposition of the end-of-life oxidation will be even lower.

Stress and Strain Limits – Pellet/Cladding Interaction

Cladding strain limits were established to protect the fuel rod cladding from failure due to Pellet/Cladding Interaction (PCI). A heavier oxide or crud layer on the fuel rod cladding would increase fuel rod temperatures leading to a faster rate of cladding creepdown and greater differential thermal expansion of the fuel pellet during a transient. As indicated above, the temperature change is negligible such that no reduction in PCI margin is expected. Based on previous evaluations, the calculated cladding strain showed essentially no dependence on the presence of a small crud layer.

Stress and Strain Limits – Cladding Stress

The steady-state cladding stress analysis is done for both beginning of life (BOL) and end of life (EOL) conditions to demonstrate cladding integrity. Stress limits were derived from the ASME Boiler and Pressure Vessel Code, Section III, Appendix III, Article III-2000. Since the potential increase in oxidation remains well within the oxidation limit (see "Oxidation, Hydriding and Crud Buildup" below), the assumed wall thinning due to corrosion in the stress calculation continues to be appropriate. No change in margin to the stress limit is indicated.

Cladding Rupture

The cladding strength and ductility during a LOCA determines the cladding to fuel pellet gap size and heat transfer rate, and these influence the overall peak cladding temperature (PCT) predicted during the transient. The evaluation of clad swell and rupture is simulated in Framatome ANP LOCA analyses for Zircaloy cladding based on the NRC evaluation presented in NUREG-0630. The analysis is performed as part of the reload licensing and is evaluated each cycle. At this time there is no known evidence that suggests the introduction of small quantities of zinc into the RCS would change the overall strength or ductility of the cladding based on any small increase in oxidation or crud buildup during the course of the cycle. Given this understanding, there is no significant effect on the swelling and rupture models used in the Appendix K LOCA analyses.

Fuel Rod Mechanical Fracturing

Fuel rod mechanical fracturing criteria protect the fuel rod from failure due to externally applied forces such as earthquakes or postulated pipe breaks. The criteria are based on stress limits derived from the ASME Boiler and Pressure Vessel Code, Section III, Appendix F for faulted conditions. The stress limits are derived using the specified BOL minimum tensile properties. The stress limits and stress calculations are not impacted by the potentially small increase in cladding oxidation and crud buildup due to zinc addition. Therefore, the margin of safety for mechanical fracturing is not affected.

Fuel Densification and Swelling

Fuel densification and swelling are limited by the design criteria for fuel temperature, cladding strain, cladding collapse and rod internal pressure criteria. The impact of the zinc addition on fuel densification and swelling is addressed by the evaluation for the other fuel rod criteria. Since the effect of zinc addition on fuel temperature is negligible, there will be no impact on the fuel densification and swelling.

Stress, Strain or Loading Limits on Assembly Components

Stress, strain and loading limits are established to prevent structural failure of fuel assembly components during normal operation, AOO, and faulted conditions. The limits are derived from the ASME Boiler and Pressure Vessel Code, Section III, Appendix III, Article III-2000 and include either stress or load limits. For the fuel rod, the stress and strain limits used to protect against PCI failure apply.

A small, potential increase in cladding oxidation and crud buildup will not affect the fuel component structural strength. No additional oxidation will occur on non-fueled components due to the extra crud buildup. Thus, no reduction in safety margin would occur if small amounts of crud were deposited on the structural elements of the fuel assembly.

Fatigue

Cladding fatigue due to operational power changes is limited to 0.9 fraction of the fatigue design life if irradiated Zircaloy. The cyclic loading on the cladding occurs principally from imposed strain increments from cyclic thermal expansion of the fuel pellets. Cyclic strain will not be affected by small changes in crud buildup or oxidation. Cladding creepdown can be impacted by increased cladding temperatures. However, the cladding temperature increase associated with the potential, added crud and oxidation is negligible such that no measurable change in creepdown rate would be expected. The margin of safety afforded by the 0.9 limit does not change.

Fretting Wear

Fretting wear is evaluated so fuel rod failures due to vibration do not occur. Fretting is evaluated by out-of-reactor flow testing at Framatome ANP in the hydraulic test facility as part of design qualification. In addition, acceptable fretting performance is based on operating experience. The potential, small increase in cladding oxidation and crud buildup will not affect the fretting performance because the operating experience at other plants with higher levels of oxidation and crud have demonstrated no adverse impact on fuel fretting. The margin of safety as related to fretting wear is not changed by the zinc addition.

Rod Bow

The effect of rod bow is considered in establishing thermal limits. Rod bow correlations are empirically derived from rod bow measurements on irradiated fuel assemblies. Lateral rod bow occurs from axial rod restraint at the spacer grids, neutron fast flux and thermal gradients, initial rod restraint, rod bow and residual stresses. The potential increase in oxidation or crud buildup is not related to known mechanisms for rod bow. Therefore, the margin of safety as related to rod bow is not affected.

Oxidation, Hydriding and Crud Buildup

Oxidation is limited by Framatome ANP design criteria to be less then 100 microns where the oxidation is calculated as a best estimate prediction of the oxide thickness for the highest burn-up fuel rod in each batch within the core. The limit was established from operating experience. Previous operating experience with non-optimized fuel rod cladding indicated successful operation with oxidation levels as high as 160 microns. In the case of zinc addition at ANO-1, a small potential increase in oxidation may occur. The maximum calculated EOL oxidation in the current analysis of record is 96 microns. As already mentioned, the increase in oxidation due to the small addition of crud is insignificant. The oxidation with a small increase in crud will still be within the limit of oxidation at which fuel rod failures occur remains unchanged.

Hydrogen concentration in the cladding is protected by the oxidation criteria. The same margin that exists for oxidation continues to exist for hydrogen concentration levels. Although there is no specific limit on crud buildup, Framatome ANP fuel performance codes include the effect of a nominal amount of crud buildup. As already mentioned, the effect of crud is small under normal water chemistry conditions. Previous evaluations show the impact of a small increase in crud level to be insignificant. The margins to safety for the thermal calculations and stress calculations are not impacted.

Axial Growth

Axial growth is limited to maintain proper clearances between interfacing components. The fuel rod and fuel assembly growth correlations were established as a function of fuel assembly burn-up. Although fuel rod axial growth can be affected by irradiation creep (radial creepdown), PCI, and possibly cladding oxidation, these effects are bounded by the conservative 95/95 limits (95% confidence that 95% of the values are within tolerance) established from rod growth data on a variety of fuel rod designs. The separate treatment of fuel rod and fuel assembly growth data for the calculation of differential growth introduces additional conservatism in the analysis. Thus, the potential for a small increase in crud and oxidation will remain bounded by the growth calculations. The margin of safety will not be affected.

Rod Internal Pressure

Fuel rod pressure is limited to prevent unstable thermal behavior and prevent fuel rod failure. The expected increase in fuel rod temperature, as based on previous evaluations, is insignificant for a small crud buildup increase. The resultant increase in rod internal pressure would also be small because of the insignificant change in temperature. Since the rod internal pressure design limits do not change with the addition of zinc, the margin of safety to rod failure due to excessive rod pressure is not impacted.

Assembly Liftoff

Fuel assembly lift due to hydraulic loads is not allowed. If the crud buildup and oxidation were sufficiently high, then the pressure loss coefficients would measurably increase leading to higher calculated hydraulic loads. For the small, potential increase in oxidation and crud levels, the impact on flow area and flow resistance would be negligible. The change in flow resistance due to oxidation would be well within the uncertainties of prediction capabilities and fabrication tolerances. Sufficient margin exists such that a small change in flow resistance would be negligible.

Fuel Assembly Handling

Fuel assembly handling criteria requires an axial load capability of greater than 4.0 times the assembly weight. This strength is provided by the fuel assembly cage components (i.e., end fittings, guide tubes and connection hardware). The structural component strength is not affected by the potential, small crud buildup. The margin of safety for fuel handling is therefore unaffected.

Cladding Embrittlement

The requirements on cladding embrittlement relate to the loss of coolant accident requirements of 10 CFR 50.46. Framatome ANP performs LOCA analyses to demonstrate compliance with the 50.46 limits (2200ºF peak cladding temperature, local oxidation, whole core hydrogen generation from corewide oxidation, cool-able geometry, and long term cooling). The first two criteria – cladding temperature and local oxidation – prevent excessive embrittlement of the cladding allowing it to maintain a cool-able geometry during and after the accident. The ability of these criteria to prevent excessive embrittlement of the cladding depends on the cladding state, particularly the hydrogen content of the cladding prior to the LOCA accident. Cladding hydrogen pickup during operation is not significantly increased by the addition of zinc to the RCS coolant, there is no impact on the effectiveness of the LOCA criteria. Slight increases in EOL hydrogen level of 15 percent or less should not be considered significant for current cycle designs. With these considerations, the models prescribed by Appendix K of 10 CFR 50 that are used in the NRC-approved Framatome BWNT LOCA EM can be used to compute the temperatures and oxidation values and demonstrate compliance with the five criteria.

Violent Expulsion of Fuel

In a reactivity-initiated severe accident, the deposition of energy in the fuel is the critical item. A large deposition could result in melting, fragmentation and rapid expansion of the fuel pellet such that pelletcladding interaction leads to cladding failure and dispersal of fuel particles within the RCS. The degree to which cladding failure is expected results from the consideration of the severity of cladding embrittlement during operation and the amount of energy deposited in the fuel during the accident. Cladding embrittlement during operation is primarily a result of hydrogen uptake. The addition of zinc to the RCS does not significantly increase the hydrogen uptake of the cladding, as a result, the embrittlement of the cladding will not be adversely affected and the cladding will maintain its ability to remain intact during the event. The NRC has established a guideline in Regulatory Guide 1.77 and the Standard Review Plan that restricts the radially-averaged energy deposition within the fuel at any axial elevation to less than 280 cal/gm. Any additional crud buildup due to zinc will have no effect on reactor kinetics and not alter the postulated energy deposition. Therefore, the addition of zinc to the RCS will not alter the consequences of a reactivity initiated severe accident.

Fuel Ballooning/Rupture

During a loss of coolant accident, the cladding swelling and burst strain can result in flow blockage. Therefore, the LOCA analysis must consider the cladding swelling and burst strain impact on the core flow. As in the discussion on cladding rupture, Framatome ANP uses the models prescribed in NUREG-0630 for Zircaloy cladding. Any additional crud buildup due to zinc will have no significant effect on the swelling and rupture models used in the Appendix K LOCA analysis. Also, any additional crud buildup due to zinc will have minimal effect compared to the flow passage dimensions because the nominal rod-to-rod spacing is three or four orders of magnitude larger than the maximum crud buildup from zinc.

The RCS pressure boundary is not impacted by the addition of low concentrations of zinc to the primary coolant loop. As previously discussed, in investigations conducted by EPRI, zinc has been shown not to negatively impact the materials that constitute the RCS pressure boundary, i.e.; stainless steel, Alloy 600 and Alloy 690.

Finally, zinc injection will have no effect on reactor building isolation because the isolation points will not be exposed to zinc in other than trace concentrations. At these concentrations, zinc is expected to have no effect. For this and the above reasons, zinc injection will not result in a fission product barrier, as described in the FSAR, being exceeded or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☒ No

BASIS:

The proposed activity will not result in a departure from the method of evaluation as currently approved and referenced in the FSAR. The effect of slightly higher crud levels has been evaluated for each mode of fuel rod failure and fuel system damage. It is shown that a small increase in crud thickness has a negligible effect on the licensing analysis. The existing methods that are used to establish the design bases and perform the safety analyses for the fuel are judged to be adequately conservative to accommodate the increase in crud level.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-003

I. OVERVIEW / SIGNATURES

Facility: ANO – Unit 1

Document Reviewed: Work Order #33175 Task #01 Clearance 1-C18-01 **Change/Rev.:**

System Designator(s)/Description: FS

Description of Proposed Change:

It is noted in the Unit 1 FSAR sections 9.8.2 and 8.3.1.1.1 that the main transformers are provided with automatic deluge systems. The deluge system protecting the normal Main "C" transformer is to be isolated and the sprinkler piping temporarily removed to allow for the replacement of the transformer. In this configuration the deluge system is not automatic in operation nor does it provide protection to the Main "C" transformer as stated in the Unit 1 FSAR. Since this condition will exist until 2007, the proposed change will be reviewed as a temporary configuration change. The deluge system protecting the normal Main "C" transformer was installed for the protection of equipment that is required by our Insurer. This deluge system is not protecting safety related equipment nor any part of the regulatory required fire water system. In support of TAP #03-1-009 Work Order #33175 Task #01 will isolate and remove the deluge piping protecting the normal Main "C" transformer. Isolation of the deluge system will be controlled as out of service by closing valve FS-31 per Operations clearance 1-C18-01. TAP #03-1-009 changed the Unit 1 Main "C" transformer to the Spare Transformer and is not in service.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-003)**	**Sections I, II, and IV required**

Preparer: Jackie L. Johnson / **ORIGINAL SIGNED BY JACKIE JOHNSON** / ENS / EP&C / 2-2-05
Name (print) / Signature / Company / Department / Date

Reviewer: Dale H. Smith / **ORIGINAL SIGNED BY DALE SMITH** / ENS / EP&C / 2-15-05
Name (print) / Signature / Company / Department / Date

OSRC: J. R. Eichenberger / **ORIGINAL SIGNED BY J. R. EICHENBERGER** / 2-17-05
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM ENS-LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	Unit 1 FSAR sections 9.8.2, 8.3.1.1.1and figure 9-16 P&ID M-219 Sh. 1 will be temporarily impacted by this activity. Since this change is temporary the Unit 1 FSAR will not be permanently revised.
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES", perform an Exemption Review per Section III OR perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM ENS-LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM ENS-LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☐ Yes
☒ No

If "yes," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. Basis

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

Question 1 The operating license, TSs or NRC Orders does not discuss the main "C" transformer deluge system.

Question 1: Response: This activity will temporarily affect the Unit 1 FSAR documents. Therefore, the Unit 1 FSAR is the only LBD document being temporarily affected by this change. The Unit 1 FSAR will be impacted because the Main "C" transformer deluge system is temporarily isolated at valve FS-31. Also, while the system is isolated and the piping removed there is no automatic deluge system protecting the transformer as described in the FSAR.

Question 2: Response: Isolation and removal of piping from the Unit 1 Main "C" Transformer deluge system does not involve a test or experiment. Therefore, this activity will not involve a test or experiment not described in the FSAR.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.5.1[5](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
Commitment Tracking LRS search was utilized to perform keyword searches 50.59 Common.	Deluge w/5 system, main w/5 transformer, fire w/5 water

LBDs/Documents reviewed manually:

Manual search performed on the following LBD's.
FHA, Unit 1 FSAR section 9.8, Unit 1 FSAR App. 9D,
Unit 2 FSAR section 9.5, Unit 2 FSAR App. 9D.

Unit 1 FSAR figure 9-16

5. Is the validity of this Review dependent on any other change?

☐ Yes
☒ No

If "YES", list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure ENS-EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure ENS-EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

(<u>NOTE</u>: This section is not applicable to Waterford 3 and may be removed from 50.59 Reviews performed for Waterford 3 proposed activities.)

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure ENS-LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: _____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions.

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   As evaluated in the FSAR the design of the firewater system is such that rupture or inadvertent operation will not jeopardize the capability of safety related equipment. This proposed activity isolates and removes the deluge piping at the Unit 1 Main "C" transformer. With the deluge system out of service a rupture or inadvertent operation cannot occur. The deluge system protecting the Unit 1 Main "C" Transformer is not credited in the Unit 1 FSAR Chapter 6 or 14/15 single failure events and it cannot initiate an accident. Therefore, this activity will not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The fire water system is designed to minimize the affect of fires and the probability of pipe ruptures or inadvertent operation that has the potential to cause loss of function to components important to safety. This activity will not affect fire protection system components protecting safety related equipment and these components will remain functional for fire fighting purposes. Both primary pumps will be capable of providing 2500 GPM via the main loop fire system header as required by the FSAR. A deluge system to protect the Unit 1 Main "C" is not currently required because it is not operational nor is it energized. Therefore, the implementation of this change will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system or component important to safety previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The Unit 1 and Unit 2 FSAR's evaluate the fire water system for line breaks, misoperation, and mitigation of the consequences of fires which could have an effect on safety related equipment. Placing the Main "C" transformer deluge system out of service by closing valve FS-31 and removal of the above ground piping will not affect other fire systems capability to perform within design requirements as evaluated for the protection of safety related equipment. There are no accidents evaluated in the FSAR that will have their radiation dose consequences altered as a result of this temporary activity. Thus, this activity will not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The fire water system is designed such that any failure will not affect equipment important to safety. The deluge system protecting the Main "C" transformer is located outside of plant structures and in an area where a failure would not affect safety related equipment. The activity proposed by this change does not affect or change the failure mode of any equipment important to safety. Therefore, removal of the Main "C" transformer deluge system from service will not result in more than a minimal increase in the consequences of a malfunction of a structure, system or component important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The FSAR evaluates the fire water system for line breaks, misoperation, mitigation and consequences of fires which could have an effect on safety related equipment. The Unit 1 Main "C" Transformer does not provide protection to any structures, systems or components important to safety. Isolation of the Main "C" transformer deluge system and removal of the piping will not affect other fire protection system's capability of providing protection to those areas having safety related equipment as evaluated in the FSAR. Therefore, this activity will not create a possibility for an accident of a different type than any previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   Isolation of the equipment and removal of deluge piping associated with this change are all located outside of plant structures and in an area such that failure would not affect any safety related equipment. The Main "C" transformer deluge system is not safety related equipment and does not have an interface with any equipment that is important to safety. The removal of Main "C" transformer from service does not modify or affect the fire protection system's interface with other systems, structures or components. Therefore, this activity will not create a possibility for a malfunction of a structure, system or component important to safety with a different result than any previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

   BASIS:

   The isolation boundary for this change is located in the yard area fire water valve pit for the main transformers. The fire water supply to both Containment Buildings will not be altered by this activity. Therefore, this activity will not affect the fuel cladding, RCS boundary or containment for either Unit 1 or Unit 2.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

   BASIS:

   This temporary activity removes the suppression capability to a transformer that is not connected to the station and in this state is not considered to be a fire hazard. No calculations or methodologies are being changed by the temporary isolation of the Main "C" Transformer deluge system. Therefore, this activity will not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or safety analysis.

If any of the above questions is checked "YES", obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure ENS-LI-113.

ANO 50.59 Evaluation Number

2005-005

I. OVERVIEW / SIGNATURES

Facility: ANO – Unit 1

Document Reviewed: SOLUBLE POISON CONCENTRATION CONTROL (1103.004) **Change/Rev.:** 017-05-0

System Designator(s)/Description: **CHEMICAL ADDITION SYSTEM**

Description of Proposed Change:

The Batch Controller (FQI-1248) and the condensate addition valve (CV-1251) for the primary chemical injection system are degraded and require maintenance action. To allow sufficient maintenance time, it is necessary to provide contingencies for lowering boron concentrations in the RCS by an alternate lineup, bypassing the Batch Controller interlocks and CV-1251. This alternate lineup can be terminated in the Control Room by closing Injection Valve CV-1250. The concern of Moderator Dilution arises due to the removal of automatic interlocks associated with the Batch Controller during the time corrective maintenance is in progress. Operator action and monitoring is relied on during this period to terminate any dilution beyond the calculated target.

Another proposed change is the re-sequencing of two steps in section 9 –Alternate Boration from BAAT, Bypassing the Batch Controller. This change makes the evolution more efficient by allowing valve manipulations to be performed in a more logical sequence, eliminating unnecessary repeat travel between floors in the Auxiliary Building. There is no impact on the intent, purpose or outcome of the evolution.

The last proposed change is to replace an obsolete graph in Attachment E with the one already specified in step 5.2.1 of this procedure. This graph correlates BAAT volume and concentration against RCS temperature. The graph replacement was overlooked when the specs (Including the graph) for soluble poison control were shifted from Tech Specs to the Technical Requirements Manual (TRM).

Check the applicable review(s): **(Only the sections indicated must be included in the Review.)**

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-005)**	**Sections I, II, and IV required**

Preparer: Tom Van Schaik / **ORIGINAL SIGNED BY TOM VAN SCHAIK** / EOI / OPS / 3-10-05
Name (print) / Signature / Company / Department / Date

Reviewer: Phillip Lea / **ORIGINAL SIGNED BY PHILLIP LEA** / EOI / OPS / 3-10-05
Name (print) / Signature / Company / Department / Date

OSRC: J. R. Eichenberger / **ORIGINAL SIGNED BY J. R. EICHENBERGER** / 3-11-05
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings and 50.59 Evaluations.]

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM ENS-LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	SAR sections 7.2.2.2, 9.1.2.6.C, 9.1.2.1, 14.1.2.4,14.5 deal with interlocks utilized to prevent unnecessary dilution. These interlocks will be eliminated (For the most part) when the batch controller is pulled for maintenance.
TS Bases	☐	☒	
Technical Requirements Manual	☒	☐	TRO 3.5.1d requires BAAT operable.
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☒	☐	1NSE0900.009.0 section 9.4.3 deals with measures incorporated in design to prevent excessive dilution.
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES", perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. <u>AND</u> initiate an LBD change in accordance with NMM ENS-LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM ENS-LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☐ **Yes**
☒ **No**

If "yes," perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change <u>AND</u> initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

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3. <u>Basis</u>

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

OL/TS/NRC Orders: Searches revealed nothing in the proposed changes that challenged the descriptions or caused assumptions in these documents to be untrue. While the system operation is broadly discussed, the specific operating information for this equipment is beyond the level of detail contained in these documents. This statement is applicable to the alternate dilution lineup change, the step sequence change, and the graph replacement in Attachment E.

SAR/1NSE0900.009: Wording in SAR sections 7.2.2.2, 9.1.2.1, 9.1.2.6, 14.1.2.4 and the Nuclear Safety Evaluation will be untrue during the period the alternate dilution lineup is performed. As these sections take credit for the interlocks associated with the batch controller to minimize potential for moderator dilution, a 50.59 evaluation will be necessary. Note: Both the NSE and FSAR credit Operator action as one of the means of terminating a moderator dilution.

Neither the graph replacement or the transposition of steps in the Alternate Boration section meet the level of detail contained in the FSAR when describing system operation. The minor sequence change in section 9 is a change in sequence of performing a valve lineup prior to initiating process flow and does not change the actual flow path. For these two changes, there is no impact.

TRM: TRO 3.5.1b/d state the BAAT and associated piping/valves remain operable. As this system has no automatic or ES function and is capable of manual re-alignment to satisfy normal operating or accident conditions, Operability is maintained.

TESTS OR EXPERIMENTS: The configuration and/or flow paths employed in the two lineup changes discussed above are clearly designed to process the fluids mentioned (DI water and boric acid) into the RCS letdown system. As they were designed as bypass or alternate flowpaths and will be utilized as such for the appropriate fluids, they only suffer from not having a previously approved operational section in this procedure. Based on this reality, the utilization of these piping paths does not constitute a test or experiment and thus is not a test or experiment contrary to the SAR.

4. **References**

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.5.1[5](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
Autonomy 50.59 Unit 1	"Batch Controller", "dilution", "Moderator Dilution", "soluble poison", Batch, "interlock AND dilution", "interlock AND control rods", "emergency boration"

LBDs/Documents reviewed manually:

SAR sections 1.4, 4.2, 7.2.2.2, 9.1.2.1, 9.1.2.6C, 9.2, 14.1.2.4, 14.5 SAR figures 9-3, 9-4, TRM 3.5.1, Figure 3.5.1-1, 1NSE0900.009.0

5. **Is the validity of this Review dependent on any other change?**

☐ **Yes**
☒ **No**

If "YES", list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure ENS-EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure ENS-EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING
(NOTE: This section is not applicable to Waterford 3 and may be removed from 50.59 Reviews performed for Waterford 3 proposed activities.)

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure ENS-LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: ____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions.

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The accident at issue is Moderator Dilution. The timeframe for the alternate lineup proposed is short, in terms of minutes, with Operators on station. The required flowrate for the dilution is < 30 gpm, thus rapid dilution is out of the question. Enhanced monitoring of critical parameters [i.e. MUT level and pressure, control rod position, and Reactor Power] in conjunction with associated annunciator alarms [i.e. MUT level, Control rod limits] enable the Control Room staff to take appropriate and timely action to terminate any excessive dilution (Credit is given in both the Nuclear Safety Evaluation 0900 and the FSAR for Operator action terminating the event). A dilution to the point of Rx trip on high pressure or temperature as analyzed in the FSAR for moderator dilution is highly unlikely in this situation. The evolution proposed is expected to be necessary in very rare occasions if at all. As a result, there is no more than a minimal increase in the frequency of occurrence of a SAR evaluated accident.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   As this procedural section is intended to allow corrective maintenance on a degraded or inoperable component by utilizing a manual bypass alignment (Equipment and piping ratings/capacities are not exceeded), it is COMPENSATING for the malfunction of components important to safety. In addition, this manual lineup can be redone to provide for manual emergency boration (Emergency Boration lineup via this system is a manual activity regardless) at short notice, so that function is unaffected. Therefore there is NO increase in the likelihood of occurrence of a malfunction of an SSC important to safety.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The consequence of moderator dilution (gone unchecked) is a slow increase in Reactor Power terminated by a Reactor trip on high pressure or temperature if at power and a reduction in shutdown margin if shutdown with no criticality occurring. This proposed change introduces nothing that changes the definition or outcome of the analyzed DBA as it provides nothing more than a < 30 gpm dilution flow rate until terminated by Operators. This results in NO increase in the consequences of a FSAR evaluated accident.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   This alternate lineup is compensating for the malfunction of automatic engineered protective features of the batch controller in order to restore these protective features to operable status in a timely fashion. Although the injection flow path can be remotely isolated from the Control Room, Operators on station during this evolution also have the ability to rapidly isolate local valves to terminate the event. It must be restated that both the NSE referenced and the FSAR credit Operator action as one means of terminating this event. Based on the limited timeframe of the alternate lineup and the low injection flow employed, the consequences can be safely assumed to be no more than minimal.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   With the proposed lineup in effect and, assuming no Operator action, the result is a steady injection of < 30 gpm DI water into the RCS via the Makeup Tank. This would, eventually, mimic the analysis discussed in SAR Table 14.1 (Abnormalities Affecting Core and Coolant Boundary) resulting in a slow Rx Power increase and ultimate Rx Trip with no Radiological consequences. The scenario does not create a setup that would result in an accident of a different type than those already evaluated in the SAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   Firstly, the piping and valves associated with the alternate lineup are not being subjected to conditions beyond their design during the performance of this evolution.

   Moderator dilution is possible from many sources but ending with the same results. In this case , the malfunction would be to leave the DI injection lineup unattended with no Operator action. A stream of < 30 gpm DI water would have the results as analyzed in FSAR section 14. Thus, there is no possibility for a different result stemming from this proposed change.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

   BASIS:

   No, the accident for this proposed change to impact is moderator dilution. Moderator dilution is terminated by Reactor Trip with no challenge to fission product barriers (FSAR section 14). Thus, no design basis limit is exceeded or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

   BASIS:

   The proposed changes do not involve evaluations or methods of evaluations but, rather, lineups to temporarily bypass malfunctioning equipment. As such, there is no departure from a method of evaluation described in the FSAR.

If any of the above questions is checked "YES", obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure ENS-LI-113.

ANO 50.59 Evaluation Number

2005-009

I. OVERVIEW / SIGNATURES

Facility: **ANO - Unit 1**

Document Reviewed: ER-ANO-2002-1078-021, ANO-1 SG/RVCH Replacement - Temporary Power Inside Reactor Building **Change/Rev.:** 100%

System Designator(s)/Description:

Description of Proposed Change:

In support of the Arkansas Nuclear One Unit 1 Steam Generator (SG) and Reactor Vessel Closure Head (RVCH) Replacement, ER-ANO-2002-1078-021, ANO-1 SG/RVCH Replacement - Temporary Power Inside Reactor Building, provides temporary electrical power for the steam generator replacement (SGR) construction activities inside the reactor building. Power is required for a temporary lifting device, hatch transfer system, welding, post weld heat treatment, cutting, miscellaneous tools and lighting. The normal plant outage power sources are not sufficient to supply these requirements.

A temporary transformer will be connected to the power cables that normally feed power to Reactor Coolant Pump (RCP) Motor P32B, to supply the required temporary power. This will be done after Operations has released RCP P32B and it is no longer required to be in service.

The transformer is equipped with a primary fused disconnect. The transformer will feed secondary switchgear. The SGR electrical loads will be supplied from these switches. The RCP cables will be disconnected from the temporary transformer and reconnected to the RCP motor following completion of the Steam Generator Replacement Outage.

Power for this temporary construction power will be obtained by using the Reactor Coolant Pump (RCP) P32B power feed. (The ANO-1 6.9KV Switchgear Bus H2 is the normal power source for RCP P32B.) ANO will disconnect the power feeds at the Motor Terminal Box. A temporary motor terminal box will be fabricated and attached to the existing motor terminal box. ANO will connect the temporary feeder cables to the existing RCP feeder cables at the temporary motor terminal box. The 6.9KV switchgear Bus H2 will be protected from over current conditions on the temporary electrical distribution system by the type 200E fused disconnect on the primary side of the temporary transformer. The fuse selected for the primary disconnect on the temporary transformer will protect all SGT and ANO equipment in the event of a fault. Two 25KVA 480V/240/120V transformers with 100A 240/120V panelboards will be utilized until the main temporary transformer can be put in service and afterwards as necessary.

All temporary cables shall be routed and secured to provide maximum protection of the cables and minimum interference with other SGR activities. Cables shall not be attached to or routed directly over any critical equipment or small bore piping/supports that are required to be operable during the applicable modes. Upon SGR completion, all temporary attachments and components (i.e. transformer, power panels, cables, etc.) will be removed and plant structures shall be returned to their as-found condition and configuration.

Installation of the temporary transformer, switchgear and cable connections to the RCP feeds for the temporary transformer may be performed in Modes 5, 6, or Defueled condition after RCPs are released by Operations. Installation of the secondary distribution panels and cables may be performed in Modes 5, 6 or Defueled condition. To address potential seismic II/I concerns, the mounting of the main temporary transformer is addressed per Calc ANO-ER-04-032, contained in ER-ANO-2002-1078-011. Other, smaller temporary equipment must be restrained per guidance provided in ANO Procedure OP-1000.024, “Control of Maintenance”. All work done in this ER is temporary and upon completion of the SGR, all equipment will be returned to as-found condition.

ER-ANO-2002-1078-021 does not adversely affect the design function of an SSC as described in the FSAR, does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR, and does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished. A 50.59 Evaluation is performed at the request of ANO Licensing.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-009)**	**Sections I, II, and IV required**

Preparer: Lee Puckett / **ORIGINAL SIGNED BY LEE PUCKETT** / EOI / SG-Project / 3-15-05
Name (print) / Signature / Company / Department / Date

Reviewer: Doug Barborek / **ORIGINAL SIGNED BY DOUG BARBOREK** / EOI / SG-Project / 3-16-05
Name (print) / Signature / Company / Department / Date

OSRC: J.R. Eichenberger / **ORIGINAL SIGNED BY J.R. EICHENBERGER** / 4-7-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	ANO-1 FSAR Figure 8-1 (ANO Drawing E-1) is temporarily affected but not permanently changed. FSAR Figure 8-1 will not be changed for this temporarily affected document (see 3. Basis) and therefore an LBD change is not required.
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ **Yes** ☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change <u>AND</u> initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. **<u>Basis</u>**

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

<u>Operating License/Technical Specifications (TSs)</u>

ER-ANO-2002-1078-021, Temporary Power Inside Reactor Building, provides electrical power for the steam generator replacement (SGR) construction activities inside the reactor building, and is part of the activity to replace the SGs, RVCH and Service Structure for Unit 1. A temporary transformer will be connected to the power cables that feed Reactor Coolant Pump (RCP) Motor P32B to supply the required power. The power source for the cables is ANO-1 6.9kv Bus 1H2, which is the normal power source for RCP P32B. The transformer will be connected after Operations has released RCP P32B and it is no longer required to be in service. The cables will be disconnected from the temporary transformer and reconnected to the RCP motor prior to completion of the Steam Generator Replacement Outage.

Installation of the temporary transformer, switchgear and cable connections to the RCP feeds for the temporary transformer may be performed in Modes 5, 6, or Defueled condition after the RCP is out of service and released by Operations. Installation of the secondary distribution panels and cables may be performed in Modes 5, 6 or Defueled condition. ANO-1 Technical Specifications note RCP operability requirements for Modes 1-4 (TSs 3.4.4-6), but do not require any RCPs to be operable in Modes 5, 6, or Defueled condition.

The LRS search and manual search of the ANO-1 TSs, found no NRC orders or TSs that were affected by this ER.

<u>FSAR</u>

The temporary electrical distribution system located inside the reactor building will be connected to and disconnected from the disconnected feeder cables for RCP P32B. RCPs and RCP motors are described in FSAR sections 4.2.2.5, 4.2.2.6, and 4.3.5. Descriptions of the onsite electric systems and onsite power systems are in FSAR sections 8.1.2 and 8.3. FSAR section 8.3.1.1.2 states that "Two 6900-volt buses, designated as 1H1 and 1H2, are provided for the operation of the four Reactor Coolant Pumps." During the SGR outage, RCP P32B will be placed out of service and 1H2 will be used to provide power to the SGT temporary transformer. After Operations releases RCP P32B, the feeder cables from the 6.9kv Bus 1H2, will be removed from RCP P32B and will be connected to the SGT temporary transformer. The cables will be reconnected to RCP P32B prior to completion of the SGR outage.

The modifications to the RCP power connections are temporary and will be performed in modes in which the RCP is not required to be in operation per ANO-1 TSs. Since these changes are temporary, a change to the facility as described in the FSAR is not required. However, ANO-1 drawing E-1, "Station Single Line Drawing", will be changed per DRN 04-3483 and DRN 04-3485 to represent the temporary modified configuration during this ER – the connection of the SGT temporary transformer versus RCP P32B. This drawing is also included in the FSAR as Figure 8-1, "Station Single Line Diagram", which is noted as "affected" but is not permanently changed. This FSAR figure will not be changed in the FSAR and will not require a LBD Change Form.

Test or Experiments Consideration

Addition of the transformer and switchgear are considered temporary. They will not be installed or inservice with the unit at power. Installation and use of the temporary transformer, switchgear, and cables does not involve any permanent modifications to the facility, does not alter any procedures or methods of operation, and is not considered a test or experiment.

This modification does not permanently change the operation or function of any system as described in the FSAR, TS, or other Licensing Basis Document, and no changes to the Technical Specifications or FSAR are needed.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LRS 50.59 – Unit 1	“temporary power”, “RCP motor”, “reactor coolant pump motor”, “electric* distrib*”, “electrical separa*”, “cable separa*”, “1H2”, “6900-volt”, “6.9kv”, ”P32B”, and “P-32B”

LBDs/Documents reviewed manually:	
ANO Unit 1 FSAR	Sections 4.1.3.3, 4.2.1, 4.2.1.2, 4.2.2.5, 4.2.2.6, 4.3.5, 4.3.5.3, 4.3.5.4, 5.3.1, 7.2.3.2.5, 7.3.2.2.2, 8.1, 8.3.1.1, 9.3.2.1, and Chapter 14
	Figures 4-1, 7-20, 7-21, and 8-1
	Tables 1-2, 4-7, and 14-11
ANO Unit 1 Technical Specifications	TSs 3.4.1-8, 3.8.1, 3.8.2, B.3.4.1-8, B.3.8.1, and B.3.8.2

5. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☐ **Yes**
☒ **No**

If “YES,” list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered “yes,” an ISFSI Review must be performed in accordance with NMM Procedure LI-112, “72.48 Review,” and attached to this Review.

Will the proposed Change being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: _____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10 CFR 50.59 Review Program Guidelines for guidance.

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.

☐ Yes
☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?

   ☐ Yes
   ☒ No

   BASIS:

   The proposed change is temporary and will disconnect and reconnect the power source to RCP motor P32B after Operations has released it from service. It will be disconnected in Modes 5, 6, and Defueled, and is not required to be operable per TSs in those modes. The RCP motor power source will be connected to a temporary transformer and associated electrical equipment to provide power for the operation of various SG and RVCH replacement equipment (a temporary lifting device, hatch transfer system, welding, post weld heat treatment, cutting, miscellaneous tools and lighting, etc.) during Modes 5, 6, and Defueled. The temporary electrical distribution system is designed to have no adverse effect on the existing systems, structures, or components (SSCs) and will not directly impact any existing equipment. Upon completion of the replacement activities, all temporary equipment will be removed, RCP motor P32B will be reconnected to its normal power source, and plant structures will be returned to their design condition and configuration.

   Most accidents evaluated in ANO-1 FSAR Chapter 14 occur while the unit is at power. The Fuel Handling Accident occurs in the spent fuel pool in the Auxiliary Building or Reactor Building, while the unit is not at power during Mode 6 – Refueling. The activities in this ER occur in the Reactor Building during shutdown, in Modes 5, 6, and Defueled. The temporary disconnection from the RCP P32B motor and use of the RCP motor power source during unit shutdown as described in the ER, would not affect any system in a way that could lead to a fuel handling accident.

   When the unit returns to operation, all connections will be in their design condition. The reconnected RCP motor and power source will perform in the same manner as before, and will not affect overall system performance. These changes will not initiate or increase the possibility of occurrence of any FSAR evaluated accident described when the unit is at power. Also, these changes do not affect any SSC in a manner that would affect the frequency of occurrence of any other accident previously evaluated in the FSAR, including a fuel handling accident.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

   ☐ Yes
   ☒ No

   BASIS:

   This change is a temporary change to utilize the normal power supply to RCP P32B after it has been taken out of service. All equipment, including RCP P32B, will be reconnected and restored to their design configuration. The temporary electrical distribution system is designed to have no adverse effect on the existing SSCs and will not directly impact any existing equipment. Post-modification testing will be performed in accordance with ANO-1 requirements and work procedures as approved by ANO. This activity will not change any operability of any plant SSC, and all SSCs will function as they had in the pre-change condition. Therefore, the changes will not result in any increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

This activity will not change the operability of any plant SSC, and all SSCs will function as they had in the prechange condition. RCPs are not required by TSs to be operable in Modes 5, 6, or Defueled, and not having RCP P32B operable during this change will not increase the consequences of an accident previously evaluated in the FSAR.

The implementation or results of these activities do not alter the consequences (increase radiological dose) of a fuel handling accident (unit off-line) or other accidents (unit at power) described in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The changes per this ER are temporary and take place during Modes 5, 6, and Defueling. RCP motor P32B is not required to be operable in those modes per TS 3.4 and the disconnection of power to RCP P32B is a temporary condition. The temporary electrical distribution system is designed to have no adverse effect on the existing SSCs and will not directly impact any existing equipment. Mode restrictions and administrative controls will be used to ensure that the activities associated with this modification, including the addition and use of the temporary equipment; will not negatively affect any SSC important to safety. All systems and components affected under this ER shall be restored to their original configurations within the tolerances of their original designs prior to a mode in which they are required to be in operation per TSs. The temporary condition and the restored condition do not increase the consequences of a malfunction of an FSAR SSC important to safety.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The changes per this ER are temporary and do not involve the addition of any new accident initiators. Locations of temporary power equipment have been chosen so as not to affect safety related structures, systems, or components. Temporary equipment will be secured in such a manner as to meet the plants seismic II/I criteria, as required. All temporary components (i.e. transformer, power panels, cables, etc.) will be removed upon completion of the SGRP.

The function of the equipment affected by this modification is not required for shutdown of the unit (during Modes 5, 6, and Defueled), mitigating radiological releases or maintaining reactor coolant pressure integrity. Mode restrictions and administrative controls will be used to ensure that the activities associated with this modification will be performed without impacting systems important to safety that are required to be operable. No new radiological release paths will be created by these activities. Therefore, the possibility for an accident of a different type than previously evaluated in the FSAR is not created.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The use of RCP P32B feeder cables for inside the reactor building SGR temporary power distribution system will have no affect on RCP P32B, as this pump is not required in Modes 5, 6 or Defueled. During the refueling outage this pump will be out of service as declared by Operations. RCP power shall be restored to the original design prior to completion of the SGR outage. The temporary electrical distribution system is designed to have no adverse effect on the existing SSCs and will not directly impact any existing equipment. Post-modification testing will be performed in accordance with ANO-1 requirements as documented in ER-ANO-2002-1078-021.

All structures, systems and components affected under this ER shall be restored to their original configurations. The temporary changes to the systems addressed will not change the design criteria of the affected systems. A new type of malfunction of equipment is not created as a result of these changes. Therefore, the changes will not create a possibility for a malfunction of a SSC important to safety with a different result than any previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?

   ☐ Yes
   ☒ No

   BASIS:

   As discussed in NEI 96-07, Rev. 1 – Guidelines for 10 CFR 50.59 Implementation, this 10 CFR 50.59 question is directed at identifying the design basis limits for fission product barriers and evaluating the proposed activity to see if such limits are exceeded or altered. Examples of such fission product barriers would be the fuel cladding, the RCS boundary, and the Reactor Building boundary. This change is a temporary change to utilize the normal power supply to RCP P32B after it has been taken out of service by Operations. All equipment, including RCP P32B, will be reconnected and restored to its original configuration. Post-modification testing will be performed in accordance with ANO-1 requirements. This activity will not permanently change any operability or functionality of any plant SSC. It is concluded that the activities contained in the scope of ER-ANO-2002-1078-021 do not involve or affect the design basis limits of any of the fission product barriers.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?

   ☐ Yes
   ☒ No

   BASIS:

   The changes are temporary and involve use of a power supply to a piece of equipment declared out of service. The affected systems and equipment will be returned to the original design configuration. There are no safety analyses, design analyses or other methods of evaluation being changed as a result of this ER.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-010

I. OVERVIEW / SIGNATURES

Facility: **ANO - Unit 1**

Document Reviewed: ER-ANO-2002-1078-007, ANO-1 SG/RVCH Replacement – Reactor Building Opening
Change/Rev.: ERCN-1

System Designator(s)/Description:

Description of Proposed Change:

Purpose

In order to provide access for removal of the Original Once Through Steam Generators (OOTSGs) and Original Reactor Vessel Closure Head (ORVCH) and for movement of the Replacement Once Through Steam Generators (ROTSGs) and Replacement Reactor Vessel Closure Head (RRVCH) into the Reactor Building, a temporary construction opening (Opening) will be created in the Reactor Building wall. When this temporary access is no longer required, the Opening will be repaired and a pressure and leak test of the Reactor Building performed.

The opening will be located due west at azimuth 270°, directly above the existing 15'-6" diameter RB equipment hatch. The bottom of the construction opening will be located at elevation 401'-6" and the opening will be 22'-8" wide and 24'-0" high. The liner plate opening is 17'-8" wide and 21'-6 3/8" high with bottom of opening at elevation 402'-7 7/8". The concrete opening is intentionally larger than the liner plate opening to facilitate liner plate welding and testing, for liner plate stiffener repairs, and to allow for existing rebar projections without interfering with movement of equipment through the liner plate opening. The liner plate opening is sized to facilitate removal and replacement of the OTSGs and the RVCHs.

The work requires the removal and re-installation of selected tendons, concrete, bonded reinforcing, and an area of the liner plate. New tendons are fabricated from new materials for reinstallation. Some of the tendon anchor-heads and shims are reused material. Access to affected vertical tendons will be made available via the Tendon Access Gallery.

The creation of the Opening may begin when the Unit enters Mode 5, subject to the restriction that 15 hours have passed since reactor shutdown. (The 15-hour period is based on the decay heat rates used in Calculation 89-E-0017-03 to determine Reactor Building temperature and pressure in the event of an LODHR event.) Creation of the opening involves the following items:

- Removal of 16 hoop and 6 vertical tendons, including degreasing, within the Opening.
- Removal of the 45" thick wall within the Opening by hydro-demolition.
- Removal of reinforcing steel.
- Removal of existing tendon sheaths within the Opening (hydro-demolition and hand)
- Removal of a section of the liner plate.
- Detensioning of 18 additional hoop tendons (9 above and 9 below the Opening), including limited degreasing.
- Detensioning of 30 additional vertical tendons (15 north and 15 south of the Opening), including degreasing.

The Opening is created by removing the tendons, concrete, reinforcing steel, tendon sheaths and a section of the liner plate. Prior to concrete removal, the tendons in the Opening area will be de-greased, removed, and detensioned. 16 hoop (21H21 through 21H28 and 31H21 through 31H28) and 6 vertical (V75 through V80) tendons require removal. One hoop tendon (21H27 or 21H28) and one vertical (V77 or V78) tendon from the 16 hoop and 6 verticals will be removed during Modes 1 to 4 as allowed by procedure OP-5220.011 and calculation 11406-014. Calculation ANO-ER-03-034 evaluated the RB for removing the vertical and hoop tendons within the Opening for LODHR pressurization and temperature.

In order to minimize the compressive pre-stress in the concrete around the Opening before the replacement concrete is placed, an additional 18 hoop tendons (21H16 - 21H20 and 31H17 – 31H20 below and 21H29 – 21H32 and 31H29 – 31H33 above) and 30 vertical tendons (V60 – V74 south and V81 – V95 north) are detensioned only after the Unit has entered Mode 6 and the refueling canal filled to ≥ 23 feet of water over the top of the core or all fuel has been removed from the Reactor Building. Calculation ANO-ER-03-034 evaluated the RB for detensioning the 30 vertical and 18 hoop tendons adjacent to the Opening for applicable loads while the Unit is defueled.

The concrete in the Opening is removed during Mode 5, Mode 6, or after the Unit has been defueled. A temporary missile barrier is provided for required equipment that could be impacted by tornado missiles while in Mode 5 or Mode 6 when the RB wall thickness is less than 15". (Note: The missile barrier will not be required if procedures reference OP-1203.025, Natural Emergencies, and OP-1502.004, Control of ANO 1 Refueling, have been revised to limit fuel handling in the Reactor Building during severe weather.) Any equipment or ductwork supported on the liner plate (within the Opening limits) inside the RB is removed prior to removing the final 15 inches of concrete wall to protect the liner plate from damage. A "defense-in-depth" approach is taken with regard to an LODHR accident. The "defense-in-depth" approach toward RB pressurization while in a degraded condition has primary and secondary defenses. The primary defense is included in the 1R19 SOPP to assure an LODHR event does not occur by providing redundant heat removal methods and paths. (The 1R19 SOPP requirements were compared to the SOPP enhancements that were made for the ANO-2 SGRP and were generally found to be bounding of the ANO-2 enhancements. The only addition that will be made to the 1R19 SOPP will be to add the availability of an additional Reactor Building Cooling Unit during the time that there is less than 23 feet of water above the fuel core in the Reactor Building.) The secondary defense is the analytical approach that provides technical justification that the liner plate will not rupture when subjected to LODHR pressures. Calculation ANO-ER-03-034 evaluated the RB for LODHR pressurization and temperature. Together the primary and secondary defenses assure that a release of radioactivity due to an LODHR will not occur.

The concrete within the Opening is removed using high pressure water (hydro-demolition). Rebar is removed after being exposed. The rebar within the Opening will be marked and cut so that it can be reinstalled in its original positions. Tendon sheathing is removed as exposed by hydro-demolition with some hand assistance. The concrete rubble is tested for contamination and disposed of accordingly.

Prior to removal, the inside of the liner plate is decontaminated and its coating is removed in the area of the cut. After all of the fuel has been removed from the Reactor Building, the steel liner plate will be cut. The steel liner plate is then moved to a designated location outside of the RB for inspection and repair/preparation activities required prior to its reinstallation.

While the liner plate is removed from the Reactor Building Opening, air flow direction at the opening will be monitored. In the event that air flow is found to be out of the Reactor Building, Chemistry will assess the quantities for discharge amounts.

Runway beams, used to facilitate horizontal motion of the steam generators out of and into the RB building, are installed and supported on the RB wall at the bottom of the Opening (see ER-ANO-2002-1078-010 & ER-ANO-2002-1078-011 for details). The loads on the RB wall from the runway beams along with all other design loads and load combinations were evaluated and found to be acceptable (Calculation ANO-ER-03-034).

Restoration of OTSG/RVCH Construction Opening

The restoration of the RB Opening begins when the ROTSGs and the RRVCH have been moved into the RB and rigging equipment and structures have been removed from the RB. Restoration of the RB Opening involves the following items:

- Reinstallation of the liner plate and its angles and channel stiffeners
- Inspection and testing of the liner plate welds
- Placing of additional new reinforcing steel
- Installation/Splicing of new tendon sheaths to the existing tendon sheaths
- Splicing of new or existing reinforcing steel to the existing reinforcing steel
- Installation of the replacement tendons in the Opening
- Installing forms and placing new concrete
- Tensioning and greasing of all affected tendons.

When the Opening is no longer needed for moving equipment or material into/out of the RB, the liner plate will be reinstalled using the liner plate lifting frame that is provided by ER-ANO-2002-1078-010. The liner plate is welded back into place, tested, and accepted prior to proceeding with additional restoration activities.

Once the liner plate is welded and the entire weld is tested and accepted, RB “Closure” capability is established and refueling (Mode 6) of the Unit may begin, provided a temporary missile barrier is installed to protect required equipment. (Note: The missile barrier will not be required if procedures reference OP-1203.025, Natural Emergencies, and OP-1502.004, Control of ANO 1 Refueling, have been revised to limit fuel handling in the Reactor Building during severe weather.) Note that Unit 1 Tech. Spec. 3.9.6 requires the refueling canal water level be maintained at 23 feet (min.) above the fuel in the reactor vessel during movement of fuel assemblies (Mode 6 fuel movement). When the refueling canal is filled to ≥ 23 feet of water above the fuel, Reactor Building pressurization due to an LOSDC is not addressed in the ER since the time to steam is more than adequate for operator action to mitigate the event using the SOPP and existing plant procedures.

Once the liner plate weld is accepted, the liner plate stiffeners are repaired, tested, and accepted. When all stiffeners are accepted, the requirement to stay in Mode 6 with ≥ 23 feet of water over the core no longer applies.

New vertical and horizontal reinforcing steel (rebar) is provided and installed at the interior face of the RB wall adjacent to the liner plate stiffeners. New reinforcing steel is provided in accordance with Specification ANO-C-519. The new rebar are provided with hooked ends. Calculation ANO-ER-03-034 evaluates the final reinforcing arrangement provided in the opening for loads and stresses for the restored structure in accordance with the Unit 1 FSAR, Section 5.2.

New pipe tendon sheaths replace the existing corrugated tendon sheaths within the Opening. Special connectors are inserted into the existing sheaths at the edge of the Opening. Replacement pipe sheaths are attached to the connector with couplings. Both the connector and coupling are sealed to prevent concrete intrusion during replacement of the concrete. The vertical and horizontal tendon sheaths are field connected. The tendon sheaths are inspected and accepted by Quality Control (QC) to be in accordance with the drawings.

Insertion of the new tendons which replace the 6 removed vertical tendons (V75 through V80) and the 16 removed hoop tendons (21H21 through 21H28 and 31H21 through 31H28) may begin once all of the replacement tendon sheaths have been installed, connected and accepted. The removed tendons will be replaced with new tendons, anchor heads, and shims in accordance with Specification ANO-C-511. The pipe tendon sheaths were evaluated in Calculation ANO-ER-03-031 and found acceptable for tendon installation prior to concrete replacement. Tendon sheaths that have tendons installed prior to the placement of concrete will be examined prior to concrete placement to ensure that no damage has occurred to the sheathing connections while pulling the tendons. However, tendons will not be installed during the period from when the concrete placement starts until after the concrete attains 1000 psi compressive strength in order to avoid damage to the “green” concrete.

Existing exterior face rebar are reinstalled on the exterior face of the RB wall using mechanical splices. Limited fusion welding will be allowed where mechanical splices cannot be made per the requirement of Specification ANO-C-519. Existing #18 rebar at the top of the Opening are either reinstalled or replaced with new # 11 rebar. Existing reinforcing that is reused shall receive a VT-1 visual examination per ASME Section XI, Subsection IWL-4210 (c). Rebar mechanical splices are tested and accepted in accordance with Specifications ANO-C-514 or ANO-C-517.

Once the concrete profile, rebar and tendon sheaths are accepted, the exterior wall forms are installed using form ties anchored to the stiffeners of the liner plate.

After the forms have been installed and accepted, the concrete is placed in one continuous placement. The placement rate is controlled to limit concrete pressures on the forms and liner plate. Replacement concrete is furnished and placed in accordance with Specification ANO-C-513. A special high early strength concrete mix is utilized to enable tendon retensioning to begin at 3 days after placement. The concrete should reach the required compressive strength of 5700 psi in less than 3 days. Concrete test cylinders will be made and tested to verify that required concrete strengths are met for various construction evolutions and design requirements. Test cylinders will also be made for testing at other times including (but not limited to) 1 day, 7 days, 28 days, and 90 days. Forms will remain in place for 3 days or until the concrete has attained a compressive strength of 5700 psi.

Tendon insertion, if not completed, may begin again once the concrete has attained a compressive strength of 1000 psi. Test cylinders will be tested no sooner than 24 hours after placement. The compressive strength is expected to be greater than 1000 psi at the time that the first cylinder is tested.

When the replaced concrete attains a compressive strength of 3000 psi (approximately 24 hours) the temporary missile barrier (if used) may be removed.

Retensioning of remaining tendons may begin at 72 hours after concrete has been replaced provided the concrete has attained a compressive strength of 5700 psi. Tendons are retensioned in a sequence so as to gradually and uniformly distribute the compressive prestress in the replaced concrete of the Opening. Tendons are tensioned to specified lock-off forces and recorded.

Required Inspections & Testing

The restored RB is subjected to a combined Integrated Leak Rate Test (ILRT) and Containment Pressure Test (CPT). The ILRT pressure meets the pressure requirements and when combined with the required inspections satisfies the complete requirements of the Containment Pressure Test (CPT), per Technical Specification Section 5.5.16 and ASME requirements under the direction of the SGT Responsible Engineer (RE) and approved by the EOI RE.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-010 Rev. 1)**	**Sections I, II, and IV required**

Preparer: Wayne R. Wasser / **ORIGINAL SIGNED BY WAYNE WASSER** / Adecco / SG-RVCH / 9-1-05
Name (print) / Signature / Company / Department / Date

Reviewer: Doyle G. Adams / **ORIGINAL SIGNED BY DOYLE ADAMS** / EOI / SG-RVCH / 9-1-05
Name (print) / Signature / Company / Department / Date

OSRC: J.N. Miller / **ORIGINAL SIGNED BY J.N. MILLER** / 9-1-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	ANO-1 FSAR Sections 5.1.8, 5.1.9.3, 5.2.1.1, 5.2.1.3, 5.2.1.5.8, 5.2.3.1, 5.2.3.3, 5.2.4 and ANO-1 FSAR Figures 5-4 and 5-18.
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☐ **Yes**
☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change <u>AND</u> initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. <u>Basis</u>

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

<u>Operating License/Technical Specifications (TSs)</u>

The ANO-1 Operating License, Technical Specifications, Technical Specification Bases, and Technical Requirements Manual were reviewed to determine the impact of the proposed creation and repair of a temporary opening in the Reactor Building wall. The Reactor Building is not discussed in the Operating License. The Reactor Building is discussed in Sections 3.6, 3.9, and 5.5 of the Technical Specifications. The Reactor Building is discussed in the Tech Spec Bases Sections B3.6. The Reactor Building is not discussed in the Technical Requirements Manual or Core Operating Limits Report. No changes to the ANO-1 Operating License, Technical Specifications, Technical Specification Bases, or Technical Requirements Manual are required.

<u>FSAR</u>

The Reactor Building is discussed in FSAR Chapter 5. Section 5.1.8 discusses applicable Reactor Building Codes and Standards. This section will be revised to include the use of swaged rebar splices, rebar welding, and liner plate repair.

Section 5.1.9.3 discusses Reactor Building materials. This section will be revised to address the 3-day strength of the concrete in the Reactor Building Opening and the replacement rebar.

Section 5.2.1.1 provides a description of the Reactor Building. A new Section 5.2.1.1.1 will be added to provide a discussion of the temporary Reactor Building Opening.

Section 5.2.1.3 discusses the materials that were used in the Reactor Building structure. A new Section 5.2.1.3.1.1 will be added to discuss the concrete used in the Reactor Building Opening. A new Section 5.2.1.3.2.1 will be added to discuss the reinforcing steel used in the Reactor Building Opening, a description of the tendon sheaths used in the Reactor Building Opening will be added to Section 5.2.1.3.3, a new Section 5.2.1.3.3.2 will discuss the replacement tendons used in the Reactor Building Opening, a new Section 5.2.1.3.4.1 will be added to discuss the removal and restoration of the liner plate, and a new Section 5.2.1.3.5.1 will be added to discuss the liner plate coating in the area of the Reactor Building Opening.

Section 5.2.1.5.8 discusses the predicted state of stress of the Reactor Building. A new Section 5.2.1.5.8.2 will be added to discuss the end-of-service-life condition of the Reactor Building Opening.

Section 5.2.3.1 discusses applicable construction Codes for the Reactor Building. This section will be revised to cite the Codes used during SG/RVCH replacement.

Section 5.2.3.3 discusses inspection and testing of materials used in the Reactor Building. Section 5.2.3.3.1 will be revised to discuss the inspection and testing of the cement and aggregate used in the Reactor Building Opening, the concrete mix, rebar splices, and rebar welding.

Section 5.2.4 describes Reactor Building testing. Section 5.2.4.1.1 will be clarified to denote "original" construction and a new Section 5.2.4.1.2 will be inserted to describe the ILRT performed following SG/RVCH replacement.

Figure 5-4 will be revised to document the modification to the Reactor Building wall during the SG/RVCH replacement Outage. Figure 5-18 will be revised to show the location of the Reactor Building Opening.

Test or Experiments Consideration

The proposed modification will create a temporary opening in the Reactor building wall above the Equipment Hatch. This area will be repaired to a condition that is structurally equivalent to the as-designed wall resulting in no changes to the design functions performed by the RB. Following completion of the repairs, leakage and pressure testing will be performed in accordance with the Technical Specifications. The proposed modification is not considered a test or experiment. The Reactor Building is a passive safety feature requiring no operator action. The proposed modification will not cause the Reactor Building to be operated in a manner that is inconsistent with the FSAR descriptions.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LRS 50.59 – Unit 1 – Autonomy	“reactor building”, “containment”, tendon*, “reinforcing bar”, “reinforcing bars”, rebar*, “reinforcing steel, concrete, ILRT*, “integrated leak rate”, “integrated leakage”, “guide 1.94”, “rg 1.94”, “N45.2.5”, “cadweld”, splic*, “liner plate”, sheath*, hatch, coat*, “loss of decay”, LODHR, DR-116, SF-45, “fuel transfer tube”, “fuel handling accident”, fha*, “88-17”, “amendment no. 184”, amendment NEAR10 “184”, flood* NEAR20 hatch*, gallery

LBDs/Documents reviewed manually:	
ANO Unit 1 SAR	Sections 5.2, Chapter 14 Chapter 5 tables and figures
ANO Unit 1 Technical Specifications	Tech Specs 3.6 and 3.9

5. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☐ **Yes**
☒ **No**

If “YES,” list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☒	☐	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☒	☐	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☒	☐	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☒	☐	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☒	☐	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☒	☐	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☒	☐	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☒	☐	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

Hydro demolition of the Reactor Building Opening will involve the temporary installation of demolition equipment including high pressure pumps, high pressure pumps powered by diesel engines, robotic hydro-lasers, debris separators, and water treatment facilities. High pressure water will be used to remove the concrete in the Reactor Building Opening area. Water is then captured by an enclosure and other capture devices and is routed to an open topped collection tank. A screen(s) in the tank segregates the coarse fragments from the fine particles. Water is pumped from the tank to portable treatment equipment via hoses or temporary piping. The coarse fragments and fine particles are periodically sucked out by a vacuum truck and disposed.

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

Portable treatment equipment is setup within 300 ft. of the opening. The treatment involves a 3-step process of settling, flocculation, and pH measurement/adjustment prior to discharge. This is accomplished via a series of three filters and one treatment tank coupled with measurement and chemical storage/addition (acid and caustic) equipment. The treatment equipment is located within the bounds of a secondary containment (liner with berms) to contain any leakage or spillage. The discharge of treated/compliant wastewater stream is via an existing storm drainage system to the circulating water discharge canal. The discharge is monitored for quality and radiation periodically to assure compliance with plant and State requirements.

Once the concrete in the Opening area is removed, the hydro demolition equipment will be dismantled and removed.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☒	☐	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☒	☐	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

The proposed modification will create a temporary opening in the Reactor Building wall at the 270 degree azimuth, directly above the Equipment Hatch. The temporary opening will measure 22'-8" wide and 24' high. The bottom of the opening will be located at elevation 401'-6". The liner plate will not be cut and removed until the unit is in the de-fueled condition. Security has the option of maintaining the Reactor Building as a vital area in accordance with the ANO-1 Security Plan or de-vitalizing the Reactor Building during the outage. The liner plate will be welded back into place prior to core re-load.

The tendon galley access hatch will be opened during de-greasing of the vertical tendons. The tendon galley access hatch cover is a security boundary. Also, the security fence around the hatch will be compromised.

D INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: _____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10 CFR 50.59 Review Program Guidelines for guidance.

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation <u>ONLY</u>? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The Reactor Building functions to mitigate the consequences of accidents previously evaluated in the FSAR. The Reactor Building is not an accident initiator. The Reactor Building Opening will be restored to meet or exceed its original design condition. Therefore, the creation and repair of a temporary opening in the Reactor Building wall does not result in any increase in the frequency of occurrence of an accident previously evaluated in the FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The response to whether ER-ANO-2002-1078-007 implementation affects the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the FSAR focuses on two fundamental areas: 1) the capability of the Reactor Building in its post-restoration configuration to reliably perform its design functions, and 2) the effects implementation activities (i.e., cutting of the opening, use, and restoration) have on the reliabilities of activity-associated SSCs.

   1) <u>Restored Reactor Building SSC Design and Reliability Performance</u>

   Implementation of ER-ANO-2002-1078-007 to create and restore the temporary Reactor Building opening results in no detrimental effect upon the FSAR described design functions of the Reactor Building.

   The design and execution of the restoration of the temporary Reactor Building opening will reestablish the affected portion of the Reactor Building to a designed equivalent condition, fully capable of meeting all associated design and performance requirements. Due to procurement availability of some components and materials involved in the restoration of the temporary Reactor Building opening, equivalent and qualified substitutes are employed as necessary. The full engineering justification and demonstration of equivalence is documented by the ER-ANO-2002-1078-007 design.

   All construction materials meet or exceed the physical property and performance characteristics of those specified in FSAR Section 5.2.1, including the coating that will be applied to the inside surface of the liner plate (i.e., side exposed to the Reactor Building atmosphere) for corrosion protection. Restoration of the opening will utilize methods equivalent or exceeding those described in FSAR Section 5.2.3.3. Of particular note, the swaged rebar couplings are demonstrated to equivalently perform as compared to the cadwelds used in the original design.

   Therefore, implementation of ER-ANO-2002-1078-007 does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the FSAR.

   2) <u>ER Implementation Effects to SSC Design and Reliability Performance</u>

   ER implementation is designed and programmed to maintain the required design functions of SSCs which are still relied upon to perform their respective design function during implementation windows including Mode 5 conditions. During the Reactor Building access opening cutting; preparation; use for handling of the OOTSG, ROTSG, ORVCH, RRVCH, and missile shield; and restoration; the Reactor Building has certain functional requirements. These are:

- To provide Reactor Building closure capability during Unit 1 shutdown fueled conditions (Technical Specifications 3.9.4 and 3.9.5, Refueling Operations). For this reason, final cutting of the Reactor Building liner plate to make the Reactor Building access opening may not commence until all of the fuel is removed from the Reactor Building. At the conclusion of the SG/RVCH Replacement outage the Reactor Building liner plate will be restored and NDE performed prior to commencement of reactor refueling.
- Once the unit enters Mode 5 and 15 hours have elapsed since reactor shutdown, work will begin to remove the tendons in the opening area and to remove the concrete in the opening area. ER-ANO-2002-1078-007 documents calculations performed to demonstrate that the liner plate with no supporting concrete or other supports present can withstand the loadings resulting from the calculated Reactor Building pressure due to a Loss of Decay Heat Removal at the beginning of the outage and at the end of the outage (Reference Calculation ANO-ER-03-034). When the refueling canal is filled to ≥ 23 feet of water above the fuel, Reactor Building pressurization due to an LOSDC is not addressed in the ER since the time to steam is more than adequate for operator action to mitigate the event using the SOPP and existing plant procedures.
- The engineered safeguards systems and components required to maintain RB closure in Modes 5 and 6must be protected against loss of function due to damage by tornado generated missiles. See FSAR Sect 5.2.2.4.8 and ULD-0-TOP-08 Sect. 3.2.1. Normally the 45" RB wall protects the safeguards systems inside the RB from damage due to external missiles. The external missiles are a 4" x 12" wood plank x 12' long traveling end on at 300 mph or a 3" diameter schedule 40 pipe x 10' long traveling end on at 100 mph. See FSAR Sect 5.2.1.2.6 and ULD-0-TOP-08 Sect. 3.3.2. FSAR Sect 5.1.5 establishes 13" of 3000 psi concrete as adequate to resist spalling of concrete from the 3" dia. pipe missile which is the governing missile. 15" of RB concrete will conservatively be used as the minimum RB wall thickness to protect required equipment inside the RB from tornado generated missiles. In Modes 5 and 6 a temporary missile barrier will be provided when less than 15" of the RB wall remains in the Opening (equivalent to removing 30" of concrete). (Note: The missile barrier will not be required if procedures reference OP-1203.025, Natural Emergencies, and OP-1502.004, Control of ANO 1 Refueling, have been revised to limit fuel handling in the Reactor Building during severe weather.)
- The Reactor Building wall is classified as a security barrier. Once the liner plate has been removed, appropriate security measures are required to maintain the integrity of the breached vital area boundary, unless de-vitalized by Plant Security. The extent of the additional security measures taken by the plant's security team will be coordinated with SGT for schedule purposes and potential impact. During and subsequent to creation of the Reactor Building access opening, appropriate temporary security measures will be taken in accordance with the applicable site procedures, to ensure that the plant site security plan requirements are maintained. These security measures are then required until the vital area boundary (liner plate) has been restored. The work is dictated by the plant's security team.
- To serve as a radiation control zone during the OTSG/RVCH replacement outage. During ER-ANO-2002-1078-007 implementation, after the liner plate is removed, a potential airborne contamination release path is created. Measures will be taken to minimize and monitor the potential release of radioactive material by covering the opening with a tarp and maintaining air flow into the Reactor Building as discussed in the ER-ANO-2002-1078-007 implementation instructions. Air flow direction will also be monitored while the tarps are open to ensure that air does not flow out of the Reactor Building.
- The Tendon Gallery is accessed from the Auxiliary Building and through the Tendon Gallery Access Hatchway. During SG/RVCH Replacement activities, access to the Auxiliary Building will be barricaded. The barrier will be removed by Security to provide emergency egress, if needed. The hatchway door (DR-116) also provides a security barrier. The hatchway and tendon gallery will be devitalized once the Auxiliary Building barrier is in place. The hatch also provides flood protection for the tendon gallery. In the event of a flood, closure is ensured by OP-1203.025 - Natural Emergencies.

Therefore, implementation of ER-ANO-2002-1078-007 does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The calculations associated with ER-ANO-2002-1078-007 demonstrate that the affected portion of the Reactor Building wall will be restored to meet or exceed the design requirements of the SAR for DBA load cases. Additionally, the overall ability of the entire Reactor Building to perform its DBA functions is shown to remain unaffected by the changes implemented by this ER. The response to Question 2 delineated all design functions of the Reactor Building, and through the analyses performed by ER-ANO-2002-1078-007, demonstrates that the Reactor Building access opening can be repaired to be equivalent to the Reactor Building wall and meet all functional requirements associated with performing its accident mitigation functions to:

- Completely enclose the reactor coolant system to minimize release of radioactive material to the environment should a failure of the reactor coolant system occur (FSAR Sections 5.2.1.1 and 14.2.2.5.
- Enclose the radiological effects of other accidents which result in postulated releases to the Reactor Building atmosphere.

During the SG/RVCH replacement outage, activities to create and later to repair the Reactor Building Opening may occur in parallel with fuel handling activities. As discussed in Section I – Overview/Signatures, limitations are placed on the Reactor Building Opening activities such as cutting and restoration of the liner plate. These limitations ensure that closure of the Reactor Building would be achievable and the consequences of a fuel handling accident would not be increased.

Therefore, implementation of ER-ANO-2002-1078-007 does not result in any increase in the consequences of an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

As discussed in the response to Question 2, the design and implementation undertaken to repair the Reactor Building by restoring the rebar and tendons, and sealing the Reactor Building access opening demonstrate equivalent design functionality and reliability of the restored Reactor Building wall. Testing will be performed to demonstrate the essentially leak tightness of the Reactor Building prior to entry into Mode 4. The Reactor Building, therefore, will continue to perform its accident mitigation functions described in the response to Question 3 with no change in its capability to control the release of radioactivity to the environment and meet the offsite dose limits of 10 CFR 100 (i.e., no failure mechanism is created which could cause the Reactor Building to perform in a manner more severe than currently assumed). Stated alternatively, the design and implementation to make this Reactor Building wall repair demonstrates the equivalency in structural performance of the restored Reactor Building wall such that failure of the Reactor Building at the point of the restored Reactor Building access opening is no more likely than failure at any other Reactor Building location.

Therefore implementation of ER-ANO-2002-1078-007 does not result in any increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The modifications to the Reactor Building do not involve the addition of any new accident initiators. The Reactor Building has been analyzed for all applicable loading conditions associated with the Reactor Building Opening construction sequence and has been demonstrated to be capable of performing its required function under all modes of plant operation during which these activities are being conducted. There are no identified interactions with SSCs inside or outside the Reactor Building during these stages. No new release paths for accidents addressed in the FSAR are will be created since the liner plate will not be cut until after defueling is complete and will be restored prior to the start of fuel re-loading.

During the implementation of this ER, a "defense-in-depth" approach is taken with regard to an LODHR accident. The "defense-in-depth" approach toward RB pressurization while in a degraded condition has primary and secondary defenses. The primary defense is included in the 1R19 SOPP to assure a LODHR event does not occur by providing redundant heat removal methods and paths. (The 1R19 SOPP requirements were compared to the SOPP enhancements that were made for the ANO-2 SGRP and were generally found to be bounding of the ANO-2 enhancements. The only addition that will be made to the 1R19 SOPP will be to add the availability of an additional Reactor Building Cooling Unit during the time that there is less than 23 feet of water above the fuel core in the Reactor Building.) The secondary defense is the analytical approach that provides technical justification that the liner plate will not rupture when subjected to LODHR pressures. Together the primary and secondary defenses assure that a release of radioactivity from LODHR will not occur.

The 45-inch Reactor Building wall normally provides protection to the fuel and safety systems inside the Reactor Building. Calculation ANO-ER-03-030 shows that 10 inches of concrete thickness is required to prevent perforation due to the design basis tornado missiles: spalling is not a concern since the inside of the Reactor Building wall has a ¼ inch steel liner plate. During the concrete removal phase, a thickness of 15 inches of concrete will be used as the minimum Reactor Building wall thickness. With less than 15 inches of concrete, a temporary missile barrier will be installed. At the backend of the outage, a minimum compressive strength of 3000 psi is conservatively required for the 45 inch replaced concrete to provide tornado missile protection or the temporary missile barrier will be in place. (Note: The missile barrier will not be required if procedures reference OP-1203.025, Natural Emergencies, and OP-1502.004, Control of ANO 1 Refueling, have been revised to limit fuel handling in the Reactor Building during severe weather.)

The Reactor Building liner plate and concrete wall at the Reactor Building access opening will be restored to the design basis condition that existed before the opening was made; therefore implementation of ER-ANO-2002- 1078-007 does not create a possibility for an accident of a different type than any previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

As discussed in the response to Question 2, the design and implementation undertaken to repair the Reactor Building by restoring the rebar and tendons, and sealing the Reactor Building access opening demonstrate equivalent design functionality and reliability of the restored Reactor Building wall. Testing will be performed to demonstrate the essentially leak tightness of the Reactor Building prior to re-entry into Mode 4. The Reactor Building, therefore, will continue to perform its accident mitigation functions described in the response to Question 3 with no change in its capability to control the release of radioactivity to the environment and meet the offsite dose limits of 10 CFR 100. As a result of the demonstrated equivalence of the restored Reactor Building access opening reliability, no failure mechanism is created which could cause the Reactor Building to perform in a manner more severe than currently assumed. Stated alternatively, the design and implementation to make this Reactor Building wall repair demonstrates the equivalency in structural performance of the restored Reactor Building wall such that failure of the Reactor Building at the point of the restored Reactor Building access opening is no more likely than failure at any other Reactor Building location.

In accordance with SPEC-ANO-A-2437, qualified coatings will be used within the Reactor Building to ensure the durability of coating materials during expected normal and abnormal in-Reactor Building conditions with the ultimate purpose of preventing migration of coating materials to the recirculation sumps during accident conditions. Such practice ensures the proper accident mitigation function of the Reactor Building recirculation sumps serving the Safety Injection System and Reactor Building Spray System, as described in FSAR Chapter 6.

Therefore, implementation of ER-ANO-2002-1078-007 does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

The fission product barrier function is provided by the fuel clad, the reactor Coolant System (RCS) pressure boundary, and the Reactor Building liner plate. Activities performed by ER-ANO-2002-1078-007 will not have an adverse impact on the fuel clad or RCS pressure boundary. The Reactor Building liner plate functions as an essentially leak tight membrane which limits the release of radioactive fission products from the Reactor Building following a postulated accident. The Reactor Building liner plate in the area of the temporary Reactor Building Opening will be cut, removed, and then re-installed.

At the front end of the SG/RVCH replacement outage, concrete will be removed from the area of the temporary Reactor Building Opening using hydro demolition. This removal technique has been shown to not cause damage to the liner plate. The liner plate will not be cut until all fuel has been removed from the Reactor Building. Once cut, the liner plate will be removed and prepared for re-installation. Restoration of the Reactor Building Opening will begin with the liner plate being welded in-place. Following non-destructive examination and vacuum box testing of the closure weld, the fission product barrier function of the Reactor Building is restored and re-fueling activities can proceed.

The Reactor Building liner plate and concrete wall at the Reactor Building access opening will be restored to the design basis condition that existed before the opening was made; therefore implementation of ER-ANO-2002-1078-007 does not result in the design basis limit for this fission product barrier as described in the FSAR being exceeded or altered. This ER will have no impact on the fuel clad or RCS pressure boundary.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

The creation and repair of a temporary construction opening in the Reactor Building wall does not result in any changes to methods described in the FSAR.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-011

I. OVERVIEW / SIGNATURES

Facility: **ANO - Unit 1**

Document Reviewed: ER-ANO-2002-1078-011, ANO-1 SG/RVCH Replacement - Rigging and Handling Inside Reactor Building **Change/Rev.:** 0

System Designator(s)/Description: Reactor Building

Description of Proposed Change:

ER-ANO-2002-1078-011, ANO-1 SG/RVCH Replacement - Rigging and Handling Inside Reactor Building provides for the design, temporary installation, and use of construction aid equipment to perform the inside the Reactor Building rigging and handling for the ANO-1 SG/RVCH Replacement Project.

The equipment provided for rigging and handling of the Original Once Through Steam Generators (OOTSGs), and the Original Reactor Vessel Closure Head (ORVCH) and its Original Service Structure (OSS) while inside the Reactor Building consists of an Inside Hatch Transfer System (IHTS), a Temporary Lifting Device (TLD) including spreader beam & lifting links, Skid System(s), RVCH Lift Structure and Downending/Upending tailing device(s). This equipment will be used to maneuver the components to the Reactor Building Construction Opening that is created through the implementation of ER-ANO-2002-1078-007, ANO-1 SG/RVCH Replacement - Reactor Building Construction Opening. In reverse manner, the equipment will be used to maneuver the Replacement OTSGs (ROTSGs) from the Reactor Building Opening into their respective position inside the SG cubicles and the Replacement RVCH (RRVCH) and its Replacement Service Structure (RSS) from the Reactor Building Opening into its position on the RVCH head stand. Horizontal transfer of the OTSG and RVCH/SS components out of and into the Reactor Building via the Reactor Building Construction Opening is accomplished by a Hatch Transfer System (HTS), which is described below. The rigging and handling techniques for the replacement heavy components used in ER-ANO-2002-1078-011 are qualified by the heavy component manufacturer. The equipment provided for rigging and handling has a direct interface with plant structures, systems or components (SSCs). All rigging and handling interface loads have been evaluated through calculations and are found to be acceptable.

The HTS consists of a structural steel system that spans between the outside and inside areas of the Reactor Building. Skid tracks are mounted to this steel, and employ hydraulically operated push-pull units along with saddles, tailing devices, and other rigging attachments to appropriately support the OTSG and RVCH components and accomplish their lateral movements. In erecting the inside IHTS portion of the HTS, its constituent rails are, upon completion of the Reactor Building Construction Opening, joined with the rails of the outside Reactor Building portion of the HTS (the Outside HTS (OHTS)) that is temporarily erected as part of ERANO-2002-1078-010, ANO-1 SG/RVCH Replacement - Rigging & Handling Outside Reactor Building. The IHTS system will be used for multiple purposes including the movement of OTSGs & RVCHs, providing support for the Reactor Cavity Decking, and providing a means of smooth transition of components and materials into and out of the Reactor Building.

The TLD will be supported by the Polar Crane (L-2) bridge girders and used for heavy component lifts. During lifts with the TLD, the Polar Crane trolley will be positioned at the opposite end of the bridge. The Polar Crane is the only plant SSC that will be affected by the TLD. Since the TLD is a temporary device, there is no configuration change to the Polar Crane; therefore, there is no effect on the basic system functions. The TLD system's main function is to raise and lower OTSGs. The lifts of the ORVCH/OSS from the Reactor Vessel to the head stand and the lift of the RRVCH/RSS from the head stand to the Reactor Vessel are consider part of normal refueling outage activities and are not included in the scope of ER-ANO-2002-1078-011. Most of these lifts will be Engineered Lifts and will require an inspection and evaluation in accordance with ASME/ANSI B30.2 to qualify the crane for each specific lift.

In addition to the equipment supplied for handling the components, the scope of ER-ANO-2002-1078-011 includes the Auxiliary Crane, a temporary power transformer skid, and the Reactor Cavity Decking.

The Auxiliary Crane will be temporarily positioned inside the Reactor Building on top of the east ends of the OTSG cavity walls after the all fuel has been removed from the Reactor Building. The support skid for the auxiliary crane provides the interface to the plant structure and relies on the plant structure for support during installation. There are no permanent effects to the structure of the plant (e.g., no concrete anchor bolts or welding) from this temporary rigging and handling equipment. The evaluation of loading interface is described in calculation ANO-ER-04-042. The Auxiliary Crane will assist in miscellaneous lifts including additional support for rigging into and out of the cavities. The Auxiliary Crane will be removed prior to re-entry into Mode 6.

After the Reactor Vessel has been defueled and water has been removed from the Refueling Canal, a temporary FME barrier will be installed over the Reactor Vessel. Additionally, concrete blocks or steel plating will be installed over the Reactor Vessel for radiation shielding purposes. The plating and FME barrier will be removed prior to re-entry into Mode 6.

Temporary Reactor Cavity Decking will then be installed over the refueling canal to serve as a working platform and to provide an additional means to prevent materials from falling into the refueling canal and the Reactor Vessel. The Reactor Cavity Decking will be supported by both the IHTS and the wall of the Refueling Canal (above the liner plate). The Reactor Cavity Decking will provide some added shielding since its location is directly above the refueling canal. With the possible exception of some of the wall supports, the Reactor Cavity Decking will be removed prior to re-entry into Mode 6.

All temporary rigging and handling systems inside the Reactor Building will require an electrical power source powered by the temporary power system installed by ER-ANO-2002-1078-021, Temporary Power Inside Reactor Building. However, the temporary power transformer skid will be positioned by ER-ANO-2002-1078-011. The transformer skid will have a direct interaction with the plant structure when attached to the structural steel at El. 424'-6" above the top of "B" OTSG cavity wall. The transformer skid will be installed and secured at El 424'-6" in order to meet site seismic II/I requirements. An evaluation of the effect on plant SSCs is found in calculation ANO-ER-04-032. There are no configuration changes since the concrete and steel are returned to the design condition. Positioning of the skid will occur in Modes 5, 6, or Defueled. Removal of the skid will occur before re-entry into Mode 4.

Activities associated with rigging and handling of the OTSGs and RVCH/SS involve temporary changes to structures inside the Reactor Building. Permanent plant modifications are anticipated to be minimal and to only include changes needed to temporarily attach rigging equipment to plant structures (e.g., drilling of bolt holes, installation of anchors, attachment of tie down lugs, welding, etc.). Anchor bolts installed by ER-ANO-2002-1078-011 will be removed or driven into the concrete and grouted. All plant structures, systems, and components (SSCs) will be restored to the design condition unless specifically approved otherwise by SGT and EOI.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-011)**	**Sections I, II, and IV required**

Preparer: Wayne R. Wasser / **ORIGINAL SIGNED BY WAYNE WASSER** / Adecco / SG-RVCH / 3-29-05
Name (print) / Signature / Company / Department / Date

Reviewer: Doyle G. Adams / **ORIGINAL SIGNED BY DOYLE ADAMS** / EOI / SG-RVCH / 4-5-05
Name (print) / Signature / Company / Department / Date

OSRC: J.R. Eichenberger / **ORIGINAL SIGNED BY J.R. EICHENBERGER** / 4-7-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	SAR Figures 1-2 thru1-7. SAR Section 9.6.1.7.1. (It should be noted that the impact on these portions of the FSAR are limited to the duration of the outage and are therefore temporary. No changes to plant LBDs will be implemented as part of this ER.)
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ **Yes** ☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change <u>AND</u> initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. **<u>Basis</u>**

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

<u>Operating License/Technical Specifications (TSs)</u>

The ANO-1 Operating License, Technical Specifications, Technical Specification Bases, and Technical Requirements Manual were reviewed to determine the impact of the proposed installation and use of temporary construction aid equipment in the Reactor Building, along with the Polar Crane bridge girders. With the exception of the Refueling Canal, affected components are not discussed in the Operating License, Technical Specifications, Tech Spec Bases, Technical Requirements Manual, or Core Operating Limits Report. (The discussion related to the Refueling Canal is limited to refueling activities which is outside the scope of this ER.) The temporary construction aid equipment is not currently installed equipment and is not discussed in the above documents. The temporary construction aid equipment does not need to be included in any licensing basis documents since this equipment will be removed prior to plant operation. Therefore, no changes to the ANO-1 Operating License, Technical Specifications, Technical Specification Bases, or Technical Requirements Manual are required.

<u>FSAR</u>

The temporary rigging and handling equipment to be installed to support replacement of the OTSGs RVCH/SSs will be located inside the Reactor Building. This equipment will be used during the SG and RVCH/SS replacement outage to provide heavy lift capabilities in the Reactor Building. During Modes 5 and 6, safe load paths as established in ANO procedures will be used to move heavy loads. During OTSG and RVCH movements, all fuel will be removed from the Reactor Building.

SAR Figures 1-2 thru 1-7 show the internals of the Reactor Building at various elevations. The lifting devices and planned changes would temporarily modify these drawings while the old and new OTSGs and RVCH/SS are being installed. In addition, SAR Section 9.6.1.7.1 discusses the 11 cranes/lifting devices that are applicable for heavy loads. Even though these additional lifting devices are temporary, they are similarly required to meet the requirements of this section. The temporary construction aid equipment does not need to be included in any licensing basis documents since this equipment will be removed prior to plant operation. Therefore, no changes to the ANO-1 FSAR are required.

<u>Test or Experiments Consideration</u>

The proposed modification will install and use temporary construction aid equipment in the Reactor Building. This equipment, along with the Polar Crane bridge girders, is needed to perform the inside the Reactor Building rigging and handling for the OOTSGs, ROTSG, ORVCH/OSS, and RRVCH/OSS. Installation and later removal of rigging and handling equipment in the Reactor Building is not considered a test or experiment. The proposed use of the Polar Crane bridge girders to lift the OOTSGs and ROTSGs is consistent with the lifts of the OOTSGs during original plant construction..

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LRS 50.59 – Unit 1 – ZyFIND	"reactor building crane", "Polar Crane", NUREG*0612, "rigging", "load handling", "150" W/5 ton*, "reactor head", "vessel closure", RVCH, headstand, "head stand", "steam generator", "steam generators", "OTSG", "SG", "L-37", "L37", "refueling canal", load W/5 test*, "B30.2", " B30.4", Crane Manufactures Association of America", CMAA*, shared W/5 system*, interrelat*, "heavy load", "heavy loads", "heavy lift", "heavy lifts", "failure mode", "failure modes", "L-2", "L2", "foreign material", FME*, "intermediate cooling water", ICW, "instrument air", IA
LRS 50.59 – Unit 1 – ZyFIND	("instrument air" OR IA) w/20 cross*, shared W/5 system*, interrelat*

LBDs/Documents reviewed manually:	
ANO Unit 1 FSAR	Chapter 14, Table 1-3
ANO Unit 2 FSAR	Table 1.2-1
ANO Unit 1 Technical Specifications	Tech Specs 3.6 and 3.9
NRC SER dated Oct 1984 (0CNA108406)	

5. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☐ **Yes**
☒ **No**

If "YES," list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: _____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10 CFR 50.59 Review Program Guidelines for guidance.

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation <u>ONLY</u>? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

ER-ANO-2002-1078-011 provides for the temporary installation and use of construction aid equipment that is required to perform the rigging and handling activities for the OTSG and RVCH/SS replacements. Activities associated with the installation and use of the rigging and handling equipment and the placement of the temporary power transformer skid involve temporary changes to inside structures of the Reactor Building. Permanent modifications are limited to the temporary attachment of the rigging equipment to Reactor Building structures. These activities will be performed with the Unit in Modes 5, 6, and Defueled.

The accidents that are discussed in the FSAR that are applicable in Modes 5 or 6 or are not mode related are: fuel loading errors (FSAR Section 14.1.2.10), fuel handling accident (FSAR Section 14.2.2.3), and waste gas tank rupture (FSAR Section 14.2.2.7). The fuel loading errors and waste gas tank rupture are unrelated to activities performed by ER-ANO-2002-1078-011. Consideration of the fuel handling accident is appropriate since heavy load handling activities may be performed in the Reactor Building during the time that fuel is being handled. Heavy load handling during Modes 5 and 6 will be limited to the existing safe load paths defined by ANO procedures. (Note that the lift of the ORVCH/OSS from the Reactor Vessel to the head stand and the lift of the RRVCH/RSS from the head stand to the Reactor Vessel are consider part of normal refueling outage activities and are included in the scope of this ER.) Thus the probability of occurrence of a fuel handling accident is not increased. While Defueled, heavy load handling will not be restricted to safe load paths because all fuel will have been removed from the Reactor Building and all safety-related systems in the Reactor Building required during the defueled condition will not be required for accident mitigation.

Therefore, none of the planned activities in this ER cause more than a minimal increase in the frequency of an accident that has been previously evaluated in the FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

Activities associated with rigging and handling of the OTSGs and RVCH/SS involve temporary changes to safety related structures inside the Reactor Building. Permanent plant modifications are minimal and only include changes needed to temporarily attach rigging equipment to plant structures (e.g., drilling of bolt holes, installation of anchors, attachment of tie down lugs, welds, etc.). All plant structures, systems, and components (SSCs) will be restored to the design condition unless specifically approved otherwise by SGT and EOI. All loads applied to permanent plant structures by temporary equipment have been evaluated and found not to exceed allowable stresses.

The temporary construction equipment that will be used to lift the OOTSGs, ROTSGs, ORVCH/OSS, and RRVCH/RSS, along with other equipment and components during the OTSG/RVCH replacement outage are classified as not important to safety. The only lifting component to be used in this ER that is important to safety is the Polar Crane (L-2). The Polar Crane bridge girders will be used to support the Temporary Lifting Device (TLD) for movement of the OOTSGs and ROTSGs and may be used for movement of the RVCHs. The end trucks will be blocked during handling of the OOTSGs and ROTSGs as was done during handling of the OOTSGs during original plant construction. All OTSG and RVCH lifts included in this ER will only occur while there is no fuel in the Reactor Building. Other heavy lifts will occur during Modes 5 and 6 but all lifts will follow established safe load paths. All lifts will be performed in accordance with SGT's QEP 10.05 which meets or exceeds the applicable requirements of NUREG-0612, ENS-MA-119, OP-1005.002, and other applicable codes and standards.

As part of the activities addressed in this ER along with normal plant refueling outage activities, several Engineered Lifts will be required using the Polar Crane. As required by ASME/ANSI B30.2, inspections will be performed on the structural, mechanical, and electrical components before and after each lift to ensure functionality and structural integrity of the Polar Crane.

A review of the mechanical and electrical systems in the Reactor Building was performed to identify any systems that could be damaged by a postulated drop of a heavy component. In the event of a large component drop with the plant defueled, two systems, Intermediate Cooling Water (ICW) and Instrument Air (IA), could affect spent fuel cooling or Unit 2's Instrument Air, respectively. Neither ICW nor IA is important to safety. In the unlikely event of an OTSG drop inside the Reactor Building impacting either of these systems, ICW can be readily isolated from outside of the Reactor Building. Likewise, if there is a decrease of ANO-1 IA System pressure, ANO-2 Procedure 2203.021 requires that the cross-connect valves 2CV-3004 and 2CV-3015 be closed to maintain ANO-2's IA System pressure. No important-to-safety systems have been identified as being required in the Reactor Building during the defueled portion of the outage. Therefore, in the unlikely event of a steam generator drop, the required safety functions of structures, systems, and components important to safety will not be adversely impacted.

Therefore, none of the planned activities will increase the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The fuel handling accident is the only accident evaluated in the FSAR that is potentially impacted by the activities performed under ER-ANO-2002-1078-011. Handling of heavy loads inside the Reactor Building while the Unit is in Modes 5 and 6 will be limited to the existing safe load paths defined by established ANO procedures. The OTSG and RVCH movements that are included in the scope of this ER are limited to the time when all fuel has been removed from the Reactor Building. Installation and later removal of rigging anchor points will not affect fuel handling activities.

Based on the fact that all OTSG and RVCH movements occur while the unit is defueled and all other heavy loads lifts occur over safe and established load paths, there are no new mitigation system failures that could affect radiation dose. Therefore, the proposed heavy load handling activities in the Reactor Building will not result in an increase in the consequences of an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The rigging systems will not change the design criteria of the interfacing plant important-to-safety SSCs. The performance of the interfacing plant important-to-safety SSCs have been evaluated and found to be acceptable.

Portions of the work associated with the installation of the rigging and handling equipment will occur in the vicinity of the safety related systems. However, the majority of this work will be implemented when the reactor is defueled. Some miscellaneous work (e.g., staging materials, installing baseplates, welding) may be conducted in Modes 5 and 6. Generally, this work will be conducted sufficiently remote from any important-to-safety SSCs such that no adverse interaction could occur and no Seismic II/I conditions will be created. Where Seismic II/I conditions exist (e.g., temporary power transformer skid during Modes 5 and 6), steps have been added to the ER to protect the required important-to-safety SSCs during all design basis events in accordance with OP-5010.008.

Heavy load handling during Modes 5 and 6 will be limited to the existing safe load paths defined by ANO procedure. Rigging and handling of the OTSGs and RVCH/SSs will occur with the Unit defueled. Therefore, the activities described in this ER do not increase the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

An OOTSG drop and an ORVCH/OSS drop inside the Reactor Building are postulated to occur to address the radiological consequences associated with a failure of the rigging and handling system. The limiting radiological consequences of these drops occur with the OOTSG drop and the associated activity is released. These events are bounded by the OOTSG drop outside the Reactor Building since activity released due to an OOTSG drop in the Reactor Building would pass through the Reactor Building Purge System. This analysis is documented in Calculation ANO-ER-04-007, "X/Q OTSG & RVCH Drop Analysis". This analysis demonstrated that the OOTSG drop accident consequences would be a small fraction of the 10 CFR 100 and GDC 19 limits. The OOTSG drop release results in a lower dose at the site boundary than the waste gas tank (WGT) rupture (FSAR Section 14.2.2.7). The exclusion area boundary (EAB) dose for the WGT rupture bounds the OOTSG drop release.

The TLD will not be installed on the Polar Crane girders until all of the fuel is removed from the Reactor Building to avoid a possible Seismic II/I condition. During Modes 5 and 6, instructions have been added to this ER to ensure that Seismic II/I conditions are avoided for required important-to-safety SSCs. Where Seismic II/I conditions exist, steps have been added to the ER to protect the required important-to-safety SSCs during all design basis events in accordance with OP-5010.008.

Since the OOTSG drop release is bounded by the WGT rupture and its consequences are bounded by the WGT rupture and precautions have been added to avoid Seismic II/I conditions for required important-to-safety SSCs during Modes 5 and 6, no accidents of a different type than previously evaluated in the FSAR are created by this ER.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

ER-ANO-2002-1078-011 installs and later removes temporary construction aid equipment that is needed to rig and handle the steam generators and reactor vessel heads during the replacement outage. The construction aid components (Inside Hatch Transfer System, Temporary Lifting Device, skid system, downending/upending device, Temporary Auxiliary Crane, Reactor Cavity Decking, and temporary power transformer skid) are not important to safety and are not permanent components that will be removed at the end of the outage. There are no Failure Modes and Effects Analyses documented in the FSAR for these components.

The Polar Crane bridge girders will be used to support the TLD. These girders were used in a similar manner during the original installation of the OOTSGs. There are no failure modes or affects analysis documented in the FSAR for the Polar Crane or the Polar Crane bridge girders.

Attachment points for the temporary rigging will necessitate drilling holes in Reactor Building interior walls and welding to the structural steel. Anchor bolts installed by ER-ANO-2002-1078-011 will be removed or driven into the concrete and the holes grouted. All welds will removed and the structural steel will be restored to is design condition. The effect on the concrete and steel structures will be minor and the overall capability of the Reactor Building to withstand design loads will not be affected.

Furthermore, since safe and established load paths will be used during Modes 5 and 6 and the OTSG and RVCH/SS moves will occur when there is no fuel in the Reactor Building, different results to the only applicable accident in the FSAR, a fuel handling accident is not considered credible.

Therefore, the planned temporary and permanent changes to be performed by ER-ANO-2002-1078-011 will not create the possibility for a malfunction with a different result than previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

During Modes 5 and 6, safe and established load paths will be used for heavy lifting and Seismic II/I conditions will be avoided or resolved in accordance with OP-5010.008 so that the possibility of the design limits of a fission product barrier are not exceeded or altered.

Installation, use, and removal of the rigging and handling equipment for removal of the OOTSGs and ORVCH and installation of the ROTSGs and RRVCH will not adversely impact the pressure retaining capability of the RB or other fission product barriers. During the time that the rigging and handling equipment is installed and in use, the Unit will be defueled, the RCS will be opened to the Reactor Building atmosphere, and a temporary construction opening will have been created in the Reactor Building wall. Therefore, the installation and use of temporary rigging and handling equipment inside the Reactor Building will not adversely impact the pressure retaining capability of fission product barriers since all fission barriers have been either removed from the Reactor Building or are not required since all fuel will be removed from the Reactor Building.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

No new evaluations are necessary to qualify the design of the SSCs discussed in the FSAR for this activity; therefore, no new methods of evaluation will be used in the temporary installation and use of rigging and handling equipment for use during the replacement of the Unit 1 OTSGs and RVCH/SS.

Therefore, ER-ANO-2002-1078-011 does not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-014

I. OVERVIEW / SIGNATURES

Facility: ANO - Unit 1

Document Reviewed: ER-ANO-2003-0366-000, ANO-1 CRDM and Service Structure HVAC Ductwork Modifications **Change/Rev.:** 0

System Designator(s)/Description: RCS, RVCH, CRDM

Description of Proposed Change:

The ANO-1 Reactor Vessel Closure Head (RVCH) and the Control Rod Drive Mechanism (CRDM) Service Structure (CRDSS) has been replaced by ER-ANO-2002-0638-000 and ER-ANO-2002-0639-000. To support this upgrade, the ventilation ductwork providing airflow to the CRDSS must be modified to interface with the replacement CRDSS. ER-ANO-2003-0366-000, "ANO-1 CRDM and Service Structure HVAC Ductwork Modifications", provides the modification and improvements for the interfacing ductwork. The ductwork included in this activity is the ductwork from each of the anchors (on the south steam generator D-ring) just before the modified 20 inch and 22 inch ductwork to the inlet into the CRDSS. This includes the plenum assembly that will be permanently attached to the CRDSS. The following modifications and improvements are addressed by this review:

- Modified supports and new additional supports will be included.
- The replacement ductwork will be 14 gauge stainless steel instead of the existing galvanized steel.
- The ductwork will connect to the plenum which is designed to be moved with the CRDSS during head removal.
- All the existing ductwork is classified as safety-related. The portion of the ductwork containing the duct relief valves back to and including the connection to the unchanged ductwork will be classified as safety related while the ductwork downstream of the sections containing the relief valves will be classified non-safety. The reclassification of the ductwork will be addressed in this review.
- The replacement ductwork will have expansion joints to accommodate thermal induced movements.
- Bolted flange connections will replace a portion of the continuous welded connections.
- There will be eight supply ducts from the plenum to the CRDSS with four of the supply ducts containing adjustable control dampers. These dampers will be set to provide proper distribution of the air flow.
- The replacement ductwork is designed to reduce the maintenance effort required for disconnection and the reconnection during refueling activities.

The new ductwork is designed to allow the equivalent air flow to the CRDSS as the existing air flow based on existing fan capacity supplied to the containment supply air plenum. All the ductwork will be designed to Seismic I requirements as identified in FSAR section 6.3.3.1, which states that "all ducts are designed and supported for a Seismic Class 1 earthquake." This requirement is also reflected in Specification No. 6600-M-52A, "Specification for Sheetmetal Ductwork, Heating, Ventilation, and Air Conditioning Systems for the Arkansas Nuclear One of Arkansas Power and Light Company".

Check the applicable review(s): **(Only the sections indicated must be included in the Review.)**

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☒	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-014)**	**Sections I, II, and IV required**

Input Provided by:	Douglas Beckner / **ORIGINAL SIGNED BY DOUG BECKNER** / AREVA / NSSS / 3-31-05
	Name (print) / Signature / Company / Department / Date
Preparer:	Jerry Howell / **ORIGINAL SIGNED BY JERRY HOWELL** / EOI / SG-RVCH / 4-1-05
	Name (print) / Signature / Company / Department / Date
Reviewer:	Randall Smith / **ORIGINAL SIGNED BY RANDALL SMITH** / Universal Personnel / SG-RVCH / 4-1-05
	Name (print) / Signature / Company / Department / Date
OSRC:	J.R. Eichenberger / **ORIGINAL SIGNED BY J.R. EICHENBERGER** / 4-11-05
	Chairman's Name (print) / Signature / Date (Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	Figure 5-7 and Figure 6-13
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR? ☐ **Yes** ☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. Basis

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.3.2 for guidance.

Although not specifically addressed in the FSAR, the primary design function of the CRDSS ventilation duct system is to provide the air flow from the Reactor Building Ventilation System to the CRDSS to remove heat from components within and supported by the CRDSS. The cooling air flows out of the top of the CRDSS and back into the containment atmosphere. This maintains the temperature of these components within their operational limits.

ER-ANO-2003-0366-000 provides and installs newly designed ventilation ductwork to provide cooling air flow paths to the newly installed CRDSS from the existing ventilation ductwork. The new ductwork will replace existing ductwork that would not be compatible with the new integrated cooling air plenum design on the replacement CRDSS. The existing ductwork will be cut and removed from the horizontal runs at EL. 416 foot and EL. 413 foot (ANO-1 Dwg. M1B-536, "ANO-1 SSS Cooling Duct Assy"). The new ductwork will be connected to existing ductwork at these severed locations and will provide the cooling air supply interface with the air plenum on the CRDSS. The air plenum is designed to remain permanently attached to the CRDSS and be moved with the CRDSS. The air plenum distributes the air from the two inlet ducts to the CRDSS through eight ducts positioned around the CRDSS. Four of the eight inlet ducts contain adjustable control dampers so that the air flow distribution to the CRDSS can be adjusted.

The following features addressed by this screening are not impacted or have a positive impact on the CRDM cooling and will be addressed in the attached 10CFR50.59 Exemption:

- The replacement ductwork material will be stainless steel (14 gauge) instead of the existing galvanized steel.
- The replacement ductwork plenum is designed to remain attached to the CRDM Service Structure when the head is removed from the Reactor Vessel.
- The replacement ductwork has an expansion joint which is designed to accommodate thermal induced movement due to plant heatup and cooldown, and providing a simpler disconnect joint for RV head removal while maintaining ductwork integrity.
- Bolted flange connections will be used in place of the welded connections for the assembly of the individual ducts. The bolted connections do not reduce the ventilation ductwork structure strength or airflow capabilities.
- A flanged duct adapter will be welded to each of the existing galvanized ducts at a designated location on the D-ring wall. These adapters will provide a bolted flange connection on one end to provide a connection joint between the existing ductwork and new stainless steel ductwork.
- The replacement ductwork design provides the ability to disconnect and reconnect the ductwork to allow removal and reinstallation of the RV head to the Reactor Vessel with significantly less personnel effort and exposure.
- The existing three duct relief valves (DRV), PSV7444, PSV7445 and PSV7446, will remain available in the new configuration. PSV7444 will remain mounted in the existing ductwork. PSV7445 and PSV7446 will be relocated to the replacement stainless steel ductwork. Function and capacity of the DRVs will not change.
- There will be eight supply ducts from the replacement plenum to the CRDSS. Four of these supply ducts will contain adjustable control dampers with the purpose of providing the capability to balance the distribution of air flow the CRDSS.

- The replacement ductwork will have a new support added for each supply duct that attaches and restrains the vertical section of each duct to the D-ring wall.
- One wall support for the replacement ductwork at EL. 416 horizontal run will be modified and attached to the existing D-ring wall embed plate. Two wall supports for the replacement ductwork at EL. 413 horizontal run will be modified and attached to the existing D-ring wall embed plate.

The following feature addressed by this screening could be interpreted as having an adverse impact on the CRDM cooling and will be addressed in the a companying 10 CFR 50.59 Evaluation:

- All of the existing ductwork is classified as Safety Related. The plenum, the ductwork from the plenum to the CRDSS and the ductwork from the plenum to the vertical connections on the elbows at EL. 416 and EL. 413 will be reclassified as non-safety related.

The function of the CRDM cooling ventilation is not specifically addressed in the FSAR, but letters from the Architectural Engineer (Bechtel Corporation) and the Nuclear Supply System vendor (Babcock & Wilcox Co.) identifies the functions to be equipment protection from overheating and personnel accessibility during limited periods. No documentation could be located as to why this section of ductwork was classified as safety related, but the function of the DRVs was identified to protect upstream ductwork and plenum. No post accident cooling function requirement for the CRDM cooling could be identified, therefore the ductwork downstream of the duct relief valves would not serve a safety related function and is not required to be classified as safety related equipment. The result of this change will not affect written text in any document listed in Question II.A.1 above nor will these documents require additional information. The ductwork configuration of the CRDM cooling system is beyond the level of detail included in the FSAR for the Reactor Building Ventilation System, but Figures 5-7 and 6-13 will have minor revisions to reflect the re-routed ventilation ductwork. This proposed activity involves no Technical Specification related components and has no impact on the Technical Specifications; thus no changes to the Technical Specifications are required.

The impact to the polar crane, reactor internals, service structure and reactor supports due to the change in weight created by the replacement ductwork and plenum is addressed in ER-ANO-2002-0639-000, “ANO-1 Service Structure Replacement”

Installation of the replacement ductwork does not involve a test or experiment not described in the FSAR, because the new ductwork configuration has been thoroughly evaluated for permanent use in the CRDM cooling system and thus would not equate to a test or experiment. The proposed activity has no effect on the conduct of plant operations as described in managerial or administrative procedures. The proposed activity would have a positive impact on maintenance and will require revision of selected maintenance procedures to take advantage of the design improvements. As a result of this review it is concluded that the only LBD document change will be a minor revision to two FSAR figures, but no change to any written information contained in any LBD.

4. <u>References</u>

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LRS 50.59 – Unit 1	Ventilation, HVAC, cooling, “relief valve”, RBHV, damper, “reactor building ventilation”, duct and ductwork

LBDs/Documents reviewed manually:

FSAR Sections 1.4.44, 5.2.6, 6.3, 7.1.3.2.4 and 14.2
FSAR Tables 1-1, 1-2, 1-5, 6-9 and 7-11A
FSAR Figures 5-7 and 6-13
Technical Specification 3.6.5
Technical Specification Bases B3.6.5

5. **Is the validity of this Review dependent on any other change?** (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☐ **Yes**
☒ **No**

If "YES," list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☒ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: ____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10 CFR 50.59 Review Program Guidelines for guidance.

The function of the CRDM ventilation ductwork is to provide cooling air flow from the Reactor Building Ventilation System to the CRDM Service Structure to maintain the CRDMs and Position Indication instrumentation within their operating limits during plant operation. The service structure cooling air exits the top of the service structure and mixes with the containment atmosphere.

This Exemption address the following attributes of the replacement ductwork:

- The replacement ductwork material will be stainless steel (14 gauge) instead of the existing galvanized steel. All of the new ductwork and the plenum assembly material are of a minimum 14 gage Type 304 stainless steel sheet metal with bolted flange joints (ANO-1 Dwg. M1B-536). The use of stainless steel ductwork to replace the galvanized ductwork reduces a possible hydrogen source in post-accident containment (ANO-1 Calculation 91-E-0047-01, "ANO-1 Containment Building Hydrogen Post-LOCA"). This provides a positive effect without reducing any plant capability.
- The replacement ductwork plenum is designed to remain attached to the CRDM Service Structure when the head is removed from the Reactor Vessel. The plenum has been designed to become an integral part of the CRDSS and once installed will remain installed (ANO-1 Dwg. M1B-536). The plenum will become part of the combined CRDSS and RVCH structure and will remain attached when the RVCH is moved. The design accommodates the permanently attached RVCH lift pendants and provides no interference to the operation of the closure head stud hoist or the installation of the radiation shielding blankets.
- The replacement ductwork has an expansion joint which is designed to accommodate thermal induced movement due to plant heatup and cooldown, providing a simpler disconnect joint for RV head removal while maintaining ductwork integrity. ANO-1 dwgs. M1B-551, Shts. 1 thru 4 and M1B-573, Shts. 1 thru 4 provide the details of the flexible expansion joint that includes the bellows and elbow to connect to the vertical ductwork. The bellows will prevent distortions of the ductwork and will prevent any air flow restrictions from developing in the ductwork during these movements. ANO-1 CALC-030366E101-02, "ANO-1 Technical Report for Service Support Structure Cooling Duct Modification Structural Evaluation" confirms the structural adequacy of the expansion joints.

- A flanged duct adapter will be welded to each of the existing galvanized ducts at a designated location on the horizontal duct run along the D-ring wall. These adapters will provide a bolted flange connection for bolting to the respective stainless steel duct flange to form the connection between the existing ductwork and the new ductwork (ANO-1 Dwgs. M1B-546, “ANO-1 Duct 22” with DRV”, M1B-547, “ANO-1 Duct 20” with DRV”, M1B-548, “ANO-1 Duct Adapter 22” and M1B-549, “ANO-1 Duct Adapter 20”). A VITON or equivalent gasket is used to seal between the flanges. This insures continuity is maintained for the supply ducts to the CRDSS cooling without impacting the capability of the DRVs to provide their safety function.
- Bolted flange connections will be used in place of the welded connections on all the new ductwork except the connection to the existing ductwork. A VITON or equivalent gasket is used to seal between the bolted flange connections ensuring minimum air leakage thus providing an insignificant impact on the cooling capability. All of the replacement ductwork and plenum have been designed and analyzed to meet the same 2 psig collapse and Seismic Class I criteria as the safety related portions of the Reactor Building Ventilation System. Therefore, the bolted connections do not reduce the ventilation ductwork structural strength or airflow capabilities.
- The replacement ductwork design provides the ability to disconnect and reconnect the ductwork to allow removal and reinstallation of the RV head to the Reactor Vessel. A single point for each duct connection for disassembly/reassembly has been designed (ANO-1 Dwgs. M1B-540, “ANO-1 Duct 22” Connector” and M1B-546, “ANO-1 Duct 20” Connector”) to minimize the time and effort by maintenance personnel. Also, the connection was designed for minimum tooling, minimum personnel effort and self containment of all connection hardware for accountability of the connecting hardware. This will enhance safety and reduce time; thus reducing personnel radiation exposure during the performance of this evolution.
- The existing three duct relief valves (DRV), PSV7444, PSV7445 and PSV7446, will remain available in the new configuration. PSV7444 will remain mounted in the existing ductwork. PSV7445 and PSV7446 will be relocated to the replacement stainless steel ductwork. The function of the DRVs installed in the Reactor Building Ventilation System air ducts is to provide equalization of the external pressure and the internal pressure in the event of a sudden rise in external pressure. This would prevent any sudden increase of post accident atmospheric pressure from crushing the ducts and restricting air flow in the system. All the current ductwork and relief valves are classified Seismic Class I and safety-related. The relief capacity of the DRVs will remain unchanged, the differential actuation pressure of the DRVs remains the same and no reduction in capability of the ducts between the DRVs and the containment supply air plenum has been introduced. Therefore the functionality of the DRVs remains unchanged.
- The replacement ductwork is designed to provide at least the same cooling air flow rate as the existing ductwork. There will be eight supply ducts from the replacement plenum to the CRDSS providing approximately the same flow area as the existing ducts. Four of these supply ducts will contain adjustable control dampers (HVD-229, HVD-230, HDV-231 and HDV-232) with the purpose of providing the capability to balance the distribution of air flow to the CRDSS. The locations of the eight new entrances are the same as the locations of the original air inlet openings on the existing CRDSS. A total of 8000 scfm will be proportioned at approximately 1000 scfm at each inlet. Computational Fluid Dynamics (CFD) analyses and pressure drop calculations have been performed to demonstrate that the air flow pass the CRDMs and PI tubes for the new duct design is effectively the same (or better) as that for the existing cooling air flow paths.
- The replacement ductwork will have a new support added for each supply duct that attaches and restrains the vertical section of each duct to the D-ring wall. The ductwork that remains attached to the plenum during RVCH movement is supported by a support frame attached to the service structure platform (ANO-1 Dwg. M1B-542, “ANO-1 Support at SSS Platform”, Sht.1 and ANO-1 Dwg. M1B-543, “ANO-1 Support At SSS Platform”, Sht. 2). The ductwork that will not be removed with the CRDSS is supported as shown on ANO-1 Dwg. M1B-550, “ANO-1 Supports at Wall”. Two new embedded plates on the D-ring wall will be used to support the vertical sections of ductwork. ANO-1 Dwg. M1B-552, “ANO-1 CRDM Ventilation Ductwork Support Anchorage” provides the requirements for the two new embed plates that support the replacement vertical CRDM cooling ductwork and ANO-1. CALC-020366E101-03, “Concrete Anchorage for New Riser Duct Support” qualifies the ductwork support anchorage. These analyses confirm the adequacy of the new duct supports and confirms that there would be no adverse impact to any SSC.

- One wall support for the replacement ductwork at EL. 416 horizontal run will be modified and attached to the existing D-ring wall embed plate. Two wall supports for the replacement ductwork at EL. 413 horizontal run will be modified and attached to the existing D-ring wall embed plates. The configuration of the modified supports is shown on ANO-1 Dwg. M1B-550, ANO-1 Supports at Wall" and structural adequacy of the supports is provided in ANO-1 CALC-030366E101-02, "ANO-1 Technical Report for Service Support Structure Cooling Duct Modification Structural Evaluation"

The design standards applied to qualify the non-safety related portions of the ductwork are the "Rectangular Industrial Duct Construction Standards", 1980 and the "Round Industrial Duct Construction Standards", Second Edition, 1999. The safety-related ductwork design was qualified to the ASME AG-1-2003, "Code on Nuclear Air and Gas Treatment". The ductwork supports were qualified to the ASME B&PV Code, Section III, Division 1 – Subsection NF, "Components Supports", 1989. Seismic Class I requirements are confirmed in ANO-1 CALC-030366E101-02, "ANO-1 Technical Report for Service Support Structure Cooling Duct Modification Structural Evaluation" and meet the requirements in AP&LC-501, "Technical Specifications for Earthquake Resistance Design of Structures and/or Components Located in the Reactor Building for Arkansas Nuclear One Unit 1 Power Plant". Therefore changing the safety classification of the identified sections would not present an adverse impact on the ability of the ducts or duct relief valves to perform their intended functions.

Pressure drop calculations have been performed for both the existing configuration and the new configuration (ANO-1 CALC-030366E101-01, "ANO-1Technical Report for Service Support Structure Cooling Duct Modification Comparative CFD Analyses"). This calculation demonstrated that with the 8000 scfm air flow that the pressure drop is equivalent to the existing configuration and therefore will provide the equivalent cooling capability.

The CFD analyses determined, that with the air flow properly distributed, the cooling capability for individual CRDMs and PI tubes will be equivalent to the capability of the existing configuration and therefore will not create local hot spots for individual CRDMs or PI tubes.

The structural analysis/qualification report (ANO-1 CALC-030366E101-02), documents that the duct and duct supports meet all the code and Seismic Class I requirements for the application. This analysis also evaluated the use of the bolted connections in place of the welded connections on the existing ductwork and confirmed that there would be no adverse impact with the bolted connections. This confirms the integrity of the ductwork itself and that there would be no adverse impact to a safety function or nonsafety function of any other SSC in the vicinity of the ductwork.

The new stainless steel ductwork will not be painted, but the carbon steel supports will be painted per ANO-1 painting specification ANO-A-2437, "Technical Specification, Furnishing, Delivery, and Application of Field Painting Inside of Containment for Arkansas Nuclear One – Units 1 and 2". Complying with ANO painting spec and meeting the structural requirements would ensure that the new ductwork will not impact the capability of the containment emergency sump.

The change in material volume and surface areas are insignificant and would have a negligible impact on ANO-1 calculations 90-E-0060-1, "Containment Net Free Volume", 91-E-0043-01 "ANO-1 Reactor Building Surface Areas for the Heat Sink and Hydrogen Generation Calcs", and 94-R-1018-01, "ANO-1 Reactor Building Volume and Internal Surface Area Criteria". Therefore, the input to any safety analysis involving the containment free volume or heat sink availability does not result in any adverse changes.

The proposed modifications to the CRDM ventilation ductwork will not change how the Reactor Building Ventilation System is operated. Maintaining the heat input from the CRDSS to the containment environment equivalent to or less than the existing configuration would provide no adverse impact on the ability of the Reactor Building Ventilation System to control the containment interior air temperature to that required by the General Design Criteria, Criterion 51 – Fracture Prevention of Containment Pressure Boundary, FSAR Section 1.4.44.

The proposed modifications have no effect on the functional design of the CRDM cooling system or method of evaluation that demonstrates how a function is accomplished. All design requirements of the existing CRDM cooling supply ducts have been integrated into the replacement ductwork configuration. The CRDM cooling system will remain capable, under the worst case conditions, of supplying sufficient air flow for removing the heat from the CRDMs and RV head region to maintain CRDM and PI Tube temperatures below the maximum operational limits. Although not a design requirement, the new configuration will allow the CRDM cooling to be utilized during a natural circulation cooldown with no impact on this capability.

With the capability of the new configuration supplying the cooling air flow equivalent or better than in the existing configuration, there would be no impact on other components within or supported by the CRDSS (i.e. RADCAL, High Point Vent valves, cables).

As the above demonstrates, there would be no adverse impact on any FSAR described function either safety or non-safety for any SSC and no change to any of the existing safety analysis, none the less it was deemed prudent due to the reclassification of a portion of the ductwork from safety related to nonsafety related to prepare a 10CFR50.59 Evaluation prior to implementation at ANO-1.

IV. 50.59 EVALUATION

<u>License Amendment Determination</u>

Does the proposed Change being evaluated represent a change to a method of evaluation <u>ONLY</u>? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.	☐ Yes ☒ No

Does the proposed Change:

1.	Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?	☐ Yes ☒ No

BASIS:

Accidents and events evaluated in the FSAR that could be impacted by this proposed activity would be:

- Rod withdrawal at rated power
- Stuck-out, stuck-in, or dropped Control Rod Accident
- Rod Ejection Accident
- Loss of Coolant Accident
- Loss of Containment Integrity.

The replacement CRDM ventilation ductwork and plenum have been evaluated and determined to meet or exceed all code and owner's requirements for fabrication, installation and testing as required by ANO-1 SPEC-ANO-M-578, "ANO-1 Reactor Vessel Closure Head Replacement Project Design and Analysis Specification for Service Support Structure Cooling Duct Modification". The preparation process used to ensure this document correctly identified the design criteria and the functionality of the CRDM ventilation, and the controls imposed during the design, fabrication and installation assures the functional requirements of the CRDM ventilation ductwork and plenum are met. ANO-1 CALC-030366E101-01, "ANO-1, Technical Report for Service Support Structure Cooling Duct Modification Comparative CFD Analyses", determined that the cooling would be equivalent or better air flow to the CRDSS and would not increase containment bulk temperature. The structural adequacy of the ductwork and plenum was confirmed by ANO-1 CALC-030366E101-02, "ANO-1 Technical Report for Service Support Structure Cooling Duct Modification Structural Evaluation". The replacement ductwork and plenum are stainless steel material, which replaces galvanized material. This provides a reduction in a possible post accident hydrogen generation source. The change in the containment heat sink and free volume has been evaluated and determined to not impact the margins used in the containment peak pressure analysis. The calculated Design Basis Accident (DBA) peak pressure value will remain unchanged and thus not increase the challenge to the containment capabilities.

The CRDM cooling ventilation is not specifically addressed in the FSAR, but letters from the Architectural Engineer (Bechtel Corporation, December 10, 1968) and the Nuclear Supply System vendor (Babcock & Wilcox Co., February 3, 1969) identified the functions to be equipment protection from overheating and personnel accessibility during limited periods. No documentation could be located as to why this section of ductwork was classified as safety related. No post accident cooling function requirement for the CRDM cooling could be identified, therefore the ductwork downstream of the duct relief valves would not serve a safety related function and should not be classified as safety related equipment. The Duct Relief Valves (DRV) located in the CRDM cooling supply ducts are classified as safety related and perform a function of preventing high post accident external pressure from collapsing the upstream ducts. A break point between safety related duct and non-safety related duct was established at the connection of the vertical riser and the elbow to the horizontal duct for each the 20 inch and 22 inch duct. This maintained the DRVs in safety related ducts. Constriction of the ducts downstream of the DRVs would not prevent the DRVs from providing the relief function for the upstream ductwork and protecting the availability of the containment supply air plenum for a post LOCA containment environment (FSAR Figure No. 5-7). Total constriction of the downstream ductwork and plenum due to differential pressure would be highly unlikely, because the duct relief valves would still provide equalization of the differential pressure and the reverse flow path through the CRDSS would remain open. Additionally, all of the replacement ductwork and plenum have been designed and analyzed to meet the same 2 psig collapse and Seismic Class I criteria as the safety related portions of the Reactor Building Ventilation System.

No new accident initiators were introduced. Therefore, the installation of the replacement CRDM ventilation configuration would not introduce an increase in the frequency of occurrence of an accident previously evaluated in the FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The proposed activity could impact the reactor building environment and/or the ability to provide correct operational control of the control rods. The replacement CRDM ventilation ductwork and plenum have been evaluated and determined to meet or exceed all code and owner's requirements for fabrication, installation and testing as required by, ANO-1 SPEC-ANO-M-578 ,"ANO-1 Reactor Vessel Closure Head Replacement Project Design and Analysis Specification for Service Support Structure Cooling Duct Modification". The preparation process used to ensure this document correctly identified the design criteria and the functionality of the CRDM ventilation and the controls imposed during the design, fabrication and installation assures the functional requirements of the CRDM ventilation ductwork and plenum are met. ANO-1 CALC-030366E101-02, "ANO-1 Technical Report for Service Support Structure Cooling Duct Modification Comparative CFD Analyses", determined that the air flow would be equivalent or better than the existing flow to the CRDSS and would not increase containment bulk temperature. The CFD also confirms that the air flow distribution will satisfy the requirements for the individual CRDMs and heat removal capabilities are not compromised. The replacement ductwork and plenum are stainless steel material, which replaces galvanized material. This provides a reduction in a possible post accident hydrogen generation source. The change in heat sink and free volume has been evaluated and determined to not adversely impact the margins used in the containment peak pressure analysis. The calculated Design Basis Accident (DBA) peak pressure value will remain unchanged and thus not increase the challenge to the containment capabilities. The structural adequacy of the ductwork and plenum was confirmed by ANO-1 CALC-030366E101-02, "ANO-1 Technical Report for Service Support Structure Cooling Duct Modification Structural Evaluation". As identified above, reclassification of a portion of the ductwork from safety related to non-safety related does not reduced the reliability or capability of the ductwork. Therefore, the installation of the replacement CRDM ventilation configuration would not increase the likelihood of occurrence of a malfunction of any safety or non-safety SSC previously evaluated in the FSAR or creates any new malfunctions.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

Accidents and events evaluated in the FSAR that could be impacted by this proposed activity would be:

- Rod withdrawal at rated power
- Stuck-out, stuck-in, or dropped Control Rod Accident
- Rod Ejection Accident
- Loss of Coolant Accident.

No new radiological sources, processes or materials have been introduced. This activity will not increase the radiological activity levels in any contaminated system or reduce the capability of any barrier limiting dose rates to onsite or offsite personnel. No new flow paths have been created. The replacement ductwork and plenum will not change how any safety function is initiated or compromise the accident mitigative capability of any SSC. Therefore, the consequences of an accident previously evaluated in the FSAR would not be increased by the installation of the replacement CRDM ventilation configuration.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

No new radiological sources, processes or materials have been introduced. This activity will not increase the radiological activity levels in any contaminated system or reduce the capability of any barrier limiting dose rates to onsite or offsite personnel. No new flow paths have been created. The functionality and capability of all safety and non-safety related structures, systems and components have been maintained. T here are no reductions to any structure, system or component qualifications to perform any required function. The replacement ductwork and plenum would not compromise the mitigative capabilities of any SSC or how any safety function is initiated. There would be no reduction in redundancy or separation of any equipment for monitoring, detecting, initiating mitigation equipment or mitigating of any off-normal occurrence or transient, therefore, the consequences of a malfunction of any safety or nonsafety SSC previously evaluated in the FSAR would not increase by the installation of the replacement CRDM ventilation configuration.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The CRDM ventilation ductwork and plenum are not initiators of any accident and replacing the cooling ductwork and plenum will not cause the CRDM ventilation to become an accident initiator. All existing accident analyses remain valid and bounding for all affected SSCs. Therefore the installation of the replacement CRDM ventilation ductwork and plenum would not create the possibility for an accident of a different type than previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

Types of malfunctions:

- Fluid leaks
- Power failure
- Loss of position indication
- Cooling flow blockage
- Cross contamination
- Erroneous indications
- Support failure
- Lack of heat removal

The replacement CRDM ventilation ductwork and plenum have been evaluated and determined to meet or exceed all code and owner’s requirements for fabrication, installation and testing. Maintaining strict adherence to the applicable code requirements and owner’s specifications provides assurance that the capability and functionality of the replacement CRDM ventilation configuration will meet or exceed the requirements of the existing configuration. This assures that any malfunction of a structure, system or component would not create a different result than any previously evaluated in the FSAR and that FSAR identified malfunctions would continue to be bounded by the existing safety analysis.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

The replacement CRDM ventilation ductwork and plenum would have no impact on any design basis limit. The replacement ductwork and plenum have been evaluated and determined to provide equivalent or better air flow than the existing ductwork. This activity will not change any primary system parameter (flow, temperature, pressure or volume) or limit. There will be no change to any parameter's established operating setpoint or the margin from any setpoint to a parameter's limiting value. The change in the containment heat sink and free volume has been evaluated and determined to have no impact on the margins used in any containment peak pressure analysis. The Design Basis Accident (DBA) peak pressure value will remain unchanged and would not decrease the margin to the reactor building's design pressure. There would be no reduction in margins to any limit and there would be no change to how any SSC is operated or controlled. Therefore no design basis limit for a fission product barrier as described in the FSAR will be exceeded or altered by the installation of the replacement CRDM ventilation configuration.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

This evaluation addresses the replacement of a portion of the control rod drive service structure ventilation supply ductwork and a new plenum installed on a replacement CRDSS. The method of evaluation for the CRDM ventilation is below the level of detail provided in the FSAR, none the less the replacement ductwork and plenum were evaluated to relevant industry standards and guidelines. The industry standards and guidelines were used to ensure the replacement ductwork is qualified and approved for the appropriate application. This provided qualification and validation for use in the reactor building, but no part of this proposed activity involved a change or revision to any method of evaluation described in the FSAR and used in establishing the design bases or in the safety analyses.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-015

I. OVERVIEW / SIGNATURES

Facility: ANO - Unit 1

Document Reviewed: ER-ANO-2002- 0638-000, **"ANO-1 Reactor Vessel Closure Head (RVCH) Replacement"** **Change/Rev.:** 0

System Designator(s)/Description: RCS, RVCH

Description of Proposed Change:

Nuclear Change (NC), ANO-1 Reactor Vessel Closure Head (RVCH) Replacement, ER-ANO-2002-0638-000, proposes to replace the RVCH at Arkansas Nuclear One – Unit 1 (ANO-1). The replacement RVCH will resolve material related concerns associated with the existing RVCH and provide a replacement Service Structure Support Skirt (SSSS) to accommodate a replacement-in-kind control rod drive service structure (CRDSS), but will not change any function performed by the RVCH. As part of the new CRDSS a new lifting assembly will be employed that will remain attached to the RVCH during plant operation. The replacement CRDSS with new lifting assembly is addressed in NC ER–ANO-2002-0639-000, "ANO-1 Service Structure Replacement". In addition to replacing the RVCH and the CRDSS, the CRDSS ventilation supply ductwork will be reconfigured to better support the refueling disassembly and re-assembly for movement of the RVCH. The new ductwork configuration is addressed in NC ER-ANO-2003-0366-000, "ANO-1 CRDM and Service Structure HVAC Ductwork Modifications".

The following are the differences between the existing RVCH and the replacement RVCH:

- The original RVCH was designed and fabricated in accordance with ASME B&PV Code Section III, 1965 Edition with Summer 1967 Addendum. The replacement RVCH is designed and fabricated to the requirements of the ASME B&PV Code, Section III, 1989 Edition, no Addenda.
- The existing RVCH dome material is SA-533, Grade B, Class 1 and the head flange material is A-508-64, Class 2 (Code Case 1332-3) with dome and flange welded together. The replacement RVCH is constructed from a one-piece SA-508, Class 3 forging.
- The replacement service structure support skirt will have nine (9) twelve (12) inch diameter inspection openings. A hinged cover is provided for each opening and will remain attached to the skirt during operation.
- The thickness of the replacement upper service structure support skirt was increased to 1.575 inches versus the existing thickness of ¾ inch.
- The cladding on the replacement RVCH primary system fluid contact areas consist of an initial first layer of 309L stainless steel followed by subsequent layers of 308L stainless steel and will be slightly thicker than on the existing RVCH. The surface finish required on the existing RVCH cladding is 250 μin. Ra. and the replacement RVCH the surface finish requirement is < 125 μin. Ra.
- The replacement CRDM nozzle split nut rings will be attached with 5/16-18 UNC-2A socket head cap screws in place of the existing 3/8 – 16 UNC-2B cap screws. The function of these cap screws is to hold the split nut rings in place until the CRDM and RADCAL are attached to the flange with the mounting hex head cap screws.
- The CRDM nozzle material was changed from the existing Inconel 600 Alloy to Inconel 690 Alloy on the replacement RVCH.
- The CRDM nozzles were assembled and attached to the replacement RVCH using UNS W86152 and UNS N06052 filler material in lieu of the UNS W86182 filler used in the construction of the original RVCH.
- The existing lower CRDSS support skirt material is SA-516 Grade 70 and the replacement lower support skirt material is SA-515 Grade 70. The bolting material between the upper and lower support skirts will be changed from SA-193 Grade B-7 to SA-540 Grade B23, Class 4 bolting.
- The RVCH insulation will be reconfigured to use shorter L-panels on the outside of the RVCH and relocating a portion of the vertical insulation to inside the service structure support skirt. The inside vertical insulation will provide removal access ports for the respective support skirt inspection ports.
- The existing L-panel and dome insulation has a rated heat loss of 120 BTU/Hr-Ft_2 and the replacement Lpanel and dome insulation has a rated heat loss of 65 BTU/Hr-Ft_2 for the L-panels and 55 BTU/Hr-Ft^2 for the dome insulation.

- The existing stainless steel bands securing the L-panel insulation will be replaced with seismic buckles and keepers to maintain structural stability of the replacement L-panels.
- The original RVCH insulation was classified as safety related and the replacement RVCH insulation will be classified as non-safety related.
- The design of the new RVCH insulation interferes with the old thermal aging capsule box and a new box was designed and installed.
- The replacement RVCH will utilize the same o-ring design for the seal between the RVCH flange and the reactor vessel flange, but will have newly designed o-ring fasteners (clips, screws, and spacers) supplied with the replacement RVCH.
- The weight of the existing RVCH is 162700 lbs. and the replacement RVCH weights 172674 lbs. The difference in these weights is combined with the total weight difference between the old configuration and the new configuration with the new CRDSS, lifting assembly and ventilation configuration, and evaluated for impact on the Reactor Coolant System in ER–ANO-2002-0639-000.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☒	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-015)**	**Sections I, II, and IV required**

Input Provided by: Douglas Beckner / **ORIGINAL SIGNED BY DOUG BECKNER** / AREVA / NSSS / 4-12-05
Name (print) / Signature / Company / Department / Date

Preparer: Jerry Howell / **ORIGINAL SIGNED BY JERRY HOWELL** / EOI / SG-RVCH / 4-12-05
Name (print) / Signature / Company / Department / Date

Reviewer: Randall Smith / **ORIGINAL SIGNED BY RANDALL SMITH** / Universal Personnel / SG-RVCH / 4-13-05
Name (print) / Signature / Company / Department / Date

OSRC: J.N. Miller / **ORIGINAL SIGNED BY J.N. MILLER** / 4-13-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	Sections 4.2.2.1, 4.3.2, 4.3.3, 16.2.7 and 16.3.7, Table 1-1, Table 4-2, Table 4-3, Table 4-9, Table 4-14, Table 4-15, Table 4-16, Table 4-17, and Figure 4-4
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ **Yes** ☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change <u>AND</u> initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. **<u>Basis</u>**
Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.3.2 for guidance.

<u>Operating License/Technical Specifications</u>

This review addresses the replacement of the RVCH and the CRDSS support skirt with components designed and fabricated to the same interfacing dimensions and functional capabilities as the existing components. This would ensure that there would be no changes in the Reactor Coolant System (RCS) volume, flow paths, flow rate or functionality of components connected to the RVCH. The original reactor vessel, reactor vessel closure head and CRDSS support skirt are safety related (Q), seismic category 1, and were designed and fabricated in accordance with Babcock & Wilcox (B&W) specifications as an ASME Section III Class "A" Vessel. The original construction code was ASME B&PV Code Section III, 1965 Edition with Summer 1967 Addendum as stated in ANO-1 FSAR Table 4-2. The replacement RVCH is designed and fabricated to the requirements of the ASME B&PV Code, Section III, 1989 Edition, no Addenda. Per ASME Section XI, 1992 Edition, IWA-4170, an item used for repair/replacement activities shall meet the applicable construction code to which the original item was constructed and the existing design requirements. IWA-4170 makes further provisions that an item to be used for replacement may meet all or portions of the requirements of later editions of the construction code or ASME Code Section III when the construction code was not Section III provided the item to be used for replacement is reconciled with the Owners requirements through the Stress Analysis Report, Design Report, or other suitable method. In addition, the mechanical interfaces, fits, and tolerances shall be compatible with system and component requirements and materials shall be compatible with installation and system requirements. This has been appropriately documented and reconciled in accordance with ASME B&PV Code Section XI in CALC-020638E101-02, "ANO-1 Replacement RV Closure Head Reconciliation".

SPEC-ANO-M-587, "Design Specification for Reactor Vessel Closure Head Replacement ANO-1" was prepared for the replacement RVCH with the requirement that the design would maintain the existing Operating License and the Technical Specifications requirements. Extensive evaluations and analyses were performed to ensure that these requirements were met; therefore there will be no change to the ANO-1 Operating License or Technical Specifications.

<u>FSAR</u>

The following sections of the ANO-1 FSAR will be updated with the applicable information for the replacement RVCH, but no function or method of performing or controlling any function will change:

- Section 4.2.2.1 will identify the one piece forging used in the manufacturing of the replacement RVCH
- Section 4.3.2 will have the existing RVCH repair information deleted
- Section 4.3.3 will have the existing RVCH repair information deleted
- Table 1-1 will identify the materials use in the replacement RVCH
- Table 4-2 will identify the ASME B&PV Code applicable to the replacement RVCH
- Table 4-3 will identify the insulation changes and weight attributed to the replacement RVCH
- Table 4-9 will identify the materials of construction of the replacement RVCH
- Table 4-14 will identify the stress intensity and allowable stress values for the replacement RVCH and CRDM nozzles.
- Table 4-15 will identify the cumulative fatigue usage factors for the replacement RVCH and CRDM nozzles and will delete the fatigue usage factor for the repaired Control Rod Housing

- Table 4-16 will identify appropriate physical properties for the replacement RVCH
- Table 4-17 will identify appropriate chemical properties for the replacement RVCH
- Figure No. 4-4, will be corrected with appropriate mark numbers and the removal of the dome-to-flange weld that will not exist with the replacement RVCH
- Section 16.2.7 will identify the use of Alloy-690 for the CRDM nozzles and other RVCH penetrations to decrease the susceptibility and consequences of Primary Water Stress Corrosion Cracking (PWSCC) on the replacement RVCH
- Section 16.3.7 identifies the acceptability of the replacement RVCH with respect to the postulated intergranular separations in the forged materials.

All changes except for the additional overall weight increase of the RV Head and Service Structure and the change in classification of RVCH insulation are considered as a neutral or positive design function change and will be discussed in the Exemption portion of this review. The overall weight increase of the RV Head and Service Structure and the change in classification of RVCH insulation are being addressed in the Evaluation portion of this review. Materials utilized are less susceptible to PWSCC, which would provide a positive impact. Confirmation that the replacement RVCH would continue to provide the pressure boundary function has been provided. By maintaining the same dimensions at interfacing connections, all interfacing components are allowed to perform all functions without change.

<u>Test or Experiments Considerations</u>

This activity involves the replacement and modification of a plant component and does not involve any test or experiments.

The replacement RVCH was fabricated within the physical dimension tolerances of the existing RVCH. This allows the CRDMs to be installed on the replacement RVCH with the same configuration, which ensures that CRDM height, axial and radial positioning will be recreated on the replacement RVCH. Therefore the control rod assembly's will provide the same nuclear controls and functionality as with the existing RVCH.

The service structure support skirt will be the same height as the existing support skirt and will therefore provide the same service structure positioning relative to height as the existing configuration. The service structure will be bolted to the replacement service structure support skirt in the same orientation as the current configuration; therefore the connections for the supporting ancillaries will not be impacted.

Extensive evaluations and analyses have been performed to ensure that the replacement RVCH will provide the form, fit and functions that are performed by the existing RVCH. This allows all affected SSC's to continue to be operated within their design basis limitations/requirements as described in the FSAR and therefore would not constitute a test or experiment.

<u>Procedures</u>

By maintaining the same dimensions at interfacing connections on the replacement RVCH, all interfacing components are allowed to be re-connected in the same manner as on the existing RVCH. This would ensure that the replacement RVCH would require no changes in the Reactor Coolant System (RCS) volume, flow paths, flow rate or functionality of components connected to the RVCH. There is no change to any parameter setpoint or how any parameter is controlled. The intent of all procedures is maintained and the activity does not involve a change to any procedure described in the FSAR or any other License Basis Document.

<u>Other LBDs Identified in Section II.A</u>

There will be no change to the TS Bases, Technical Requirements Manual, Core Operating Limits Report, Offsite Dose Calculations Manual or NRC Safety Evaluation Reports due to the RVCH replacement.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LRS 50.59 – Unit 1	Cladding, clad, “reactor vessel”, “closure head”, seal, stud, o-ring, “control rod”, flange, CRDM, nozzle, ASME, adapter, RT_{NDT}, and accident

LBDs/Documents reviewed manually:

Arkansas Nuclear One Unit 1 Technical Specifications and Bases 2.1, 3.1, 3.4, 3.6, & 5.5.8

Technical Requirements Manual Revision: 6, Arkansas Nuclear One, Unit No. 1 TRM 3.4 & 4.2

Arkansas Nuclear One Unit 1 FSAR Section 1.4, 3.1.2.4.3, 3.2, 4.0, 5.1, 5.2, 6.3, 6.6, 7.2.2, 9.3.2.1, 9.6, 14.1, 14.2, 16.2 & 16.3
Chapter 1 Tables
Chapter 4 Tables
Chapter 9 Tables
Chapter 14 Tables
Chapter 3 Figures
Chapter 4 Figures
Chapter 5 Figures
Chapter 9 Figures

5. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☐ **Yes**
☒ **No**

If “YES,” list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☒ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: ____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10 CFR 50.59 Review Program Guidelines for guidance.

The proposed activity does not adversely affect the design function of an SSC as described in the SAR.

This review addresses the replacement of the RVCH and the CRDSS support skirt at ANO-1. The RVCH and support skirt provide the following design functions:

- CRD nozzle maintains the ejected rod trajectory to prevent adverse impact on adjacent assemblies
- Provides alignment for CRDM lead screw during insertion and withdrawal
- Provides the interface to connect CRDMs to the RVCH
- Provides support for the CRDMs
- Provides support for the CRDSS
- Maintains primary pressure boundary
- Provides barrier to the release of fission products
- Provides a removable seal to the reactor vessel flange
- Allows access to reactor vessel internals
- Provides flow path for reactor vessel high point vent
- Provides access to perform ASME Code Section XI inspections
- Remains functional during and after a design bases earthquake
- Provides support for the Thermal Aging Capsule Box Assembly

The differences between the original RVCH and the replacement RVCH were determined as follows:

- The original RVCH was designed and fabricated in accordance with ASME B&PV Code Section III, 1965 Edition with Summer 1967 Addendum. The replacement RVCH is designed and fabricated to the requirements of the ASME B&PV Code, Section III, 1989 Edition, no Addenda. As part of the Code reconciliation, the stress criteria were determined to be the same in both code editions with minor exceptions that made the 1989 Edition more conservative.

- The existing RVCH dome material is SA-533, Grade B, Class 1 and the head flange material is A-508-64, Class 2 (Code Case 1332-3). The replacement RVCH is constructed from a one-piece SA-508, Class 3 forging, thereby eliminating the circumferential butt weld and the formed plate dome. This removes the ASME Section XI Inservice Inspection requirement for the closure head to closure flange weld, but maintains the structural capability and functionality of the RVCH. The impact properties of this material meet the requirements of the original design specification and therefore would not affect current pressure/temperature limitations imposed on the reactor vessel. SA-508 Class 3 has other physical properties consistent with SA 508 Class 2 and was utilized because of its reduced carbon and chromium contents which makes the Class 3 less susceptible to cold cracking and underclad cracking than Class 2.
- The CRDM nozzles in the replacement RVCH are constructed from SB-167 UNS N06690 (Alloy 690) and SA-182 F304. The CRDM nozzles in the original head were constructed from SB-167 Alloy 600 in conjunction with Code Class 1336 and SA-182 F304. The change from Alloy 600 to Alloy 690 was made because of Alloy 690's demonstrated superior resistance to PWSCC. Alloy 690 has similar mechanical properties to those of Alloy 600.
- The weld filler materials were also determined to be equivalent and acceptable for ASME Code Section III code year editions 1989 and 1998.
- The replacement service structure support skirt will have nine (9) twelve (12) inch diameter inspection openings. The 12-inch holes provide access to the exterior of the reactor vessel head for inspection. A cover hatch is attached with a hinge assembly and a threaded rod. An analysis was performed to show that the cover hatch would remain intact during reactor operation and postulated seismic and/or LOCA events. Therefore the port covers would have no impact on the structural integrity of the support skirt or functionality of any other SSC and will improve the ability to inspect the inside area of the RVCH.
- The thickness of the replacement upper service structure support skirt was increased to 1.575 inches versus the existing thickness of ¾ inch. The increased thickness of the upper support skirt has no impact on its' functionality: however, it will provide additional radiological shielding to plant personnel. "ANO-1 SSS Support Skirt Analysis", demonstrates that the service support skirt and segments meet the applicable primary stress requirements of the ASME Code, Section III, 1989 Edition w/o Addendum and are therefore qualified to perform all required design basis functions.
- The interior of the existing RVCH is clad with stainless steel with a surface finish of 250 microinches. The cladding on the interior of the replacement RVCH consist of an initial first layer of 309L stainless steel followed by subsequent layers of 308L stainless steel with a finish of 125 microinches. Since the replacement RVCH final internal volume is within the manufacturing tolerances of the original RVCH, any analysis using the internal volume would remain valid and the replacement RVCH cladding would provide the equivalent capability of the existing RVCH cladding.
- The existing CRDM nozzles utilize 3/8 – 16 UNC-2B cap screws to attached the split nut rings to the bottom of the CRDM nozzle flange. The replacement split nut rings will be attached with 5/16-18 UNC-2A socket head cap screws. The function of these cap screws is to hold the split nut rings in place until the CRDM and RADCAL are attached to the flange with the mounting hex head cap screws. The structural capability and functionality of the CRDM nozzle adapters remain the same.
- The existing lower CRDSS support skirt material is SA-516 Grade 70 and the replacement lower support skirt material is SA-515 Grade 70. The bolting material between the upper and lower support skirts will be changed from SA-193 Grade B-7 to SA-540 Grade B23, Class 4 bolting. Neither of these material differences will affect the functionality or capability of the CRDSS support skirt.
- The insulation L-panels will be replaced with L-panels that are shorter and the number of L-panels will be increased from 16 to 30 to ensure the individual panel weight will be equal to or less than 100 lbs. each. To account for the area not covered by the shorter L-panels, insulation will be installed inside the service structure support. Reducing the size and weight of each panel will make handling the panels safer and easier during installation and removal without a loss of capability. The new insulation configuration will allow access to the inspection ports for RVCH top head inspection without the removal of the L-panels. The existing stainless steel bands will be replaced with seismic buckles and keepers to maintain structural stability. This will also decrease the time and effort needed to install and remove the installation. The structural adequacy of the RVCH insulation was confirmed by analysis.

- The existing L-panel and dome insulation has a rated heat loss of 120 BTU/Hr-Ft^2 and the replacement L-panel and dome insulation has a rated heat loss of 65 BTU/Hr-Ft^2 for the L-panels and 55 BTU/Hr-Ft^2 for the dome insulation. Analysis confirms that the average heat loss is below the limit of 55 BTU/Hr-Ft^2 and 65 BTU/Hr-Ft^2. This will provide an enhanced capability for heat retention in the primary system and reduce the heat input to components within the CRDSS and the containment building atmosphere. The enhanced insulation capability would have an insignificant impact on the primary heat balance. This improved insulation will reduce the RV head cooldown rate during a natural circulation cooldown. The RV head cooldown has a very minor impact on the RCS ambient losses and the RCS ambient losses are a minor contributor to heat removal during natural circulation cooldown. The cooldown rate value identified in OP-1203.013, "Natural Circulation Cooldown" is very conservative and would envelope the cooldown rate with the new insulation. The replacement RVCH insulation would have no impact on the RCS natural circulation cooldowns.
- The original RVCH insulation was classified as safety related (Q) and the replacement RVCH insulation will be classified as augmented quality. ER-ANO-2003-0954-000, "Downgrade to Augmented Quality Status of Thermal Insulation in the ANO-1 Reactor Building" provided the requalification of the RVCH insulation for a non-safety related status.
- The new insulation design would interfere with the design of the old Thermal Aging Capsule Box; therefore a replacement box was designed that provides the same loading capacity. The new box for seismic II/I qualified with the applied loads of Dead-Weight and Safe-Shutdown Earthquake (SSE). The new Thermal Aging Capsule Box will perform the same function as the old box without creating adverse affects on any SSC's design basis function.
- The replacement RVCH will utilize the same o-ring design for the seal between the RVCH flange and the reactor vessel flange. New designed o-ring fasteners (clips, screws, and spacers) will be supplied with the replacement RVCH. The replacement fasteners perform the same function as the existing fasteners. The differences between the existing and new fasteners were evaluated and determined to have no impact on the sealing function of the o-rings. This ensures that the design functions of sealing and access for refueling will be maintained the same as with the existing RVCH.

The following evaluations and analyses confirms the qualification and adequacy of the replacement RVCH and CRDSS support skirt to perform all the identified functions of the existing RVCH and CRDSS support skirt:

- CALC-020638E101-03, "ANO-1 RVCH Code Sizing Calculation", analytically demonstrates that the replacement closure head meets the ASME B&PV Code criteria for:

  a) minimum required pressure thickness of the RVCH and various attached nozzles (CRDM)
  b) minimum required reinforcement area for various openings (for CRDM nozzles) in the RVCH.

  Therefore the replacement RVCH and CRDM nozzles meet the thickness and reinforcement requirements of ASME B&PV Code, Section III, 1989 Edition with no addenda. This calculation also analytically confirmed that all three lifting lugs met the requirements specified in design specification SPEC-ANO-M-587, "Design Specification for Reactor Vessel Closure Head Replacement ANO-1".

- CALC-020638E101-07, "ANO-1 RVCH CRDM Nozzle Connection", documented the qualification to the stress and fatigue requirements of the 1989 Edition of the ASME Section III Code using the ANO-1 loads for the following items:

  - Sixty-nine (69) CRDM Housing Nozzles (Alloy 690 portion)
  - J-Groove weld (Alloy 690)
  - RV head portion surrounding the CRDM nozzle and J-Groove weld (SA-508, Class 3)
  - CRDM housing body-to-adapter weld
  - CRDM housing adapter

  Based on the loads and cycles from AREVA Doc. 18-1173987-04, "Functional Specification for Reactor Coolant System for ANO-1", the fatigue analysis indicates that the usage factor for 60 years of operation is 0.663 (< 1.0 maximum allowed by the design code) for the 690 Alloy material. The analysis identified that the usage factor for 60 years of operation for all the materials to be lower than the code allowable, which is < 1.0 maximum.

- CALC-020638E101-05, “Reconciliation of ANO-1 RVCH CRDM Nozzle Material and Weld Filler Material”, reconciled the material used to construct the CRDM nozzle housings in the ANO-1 replacement RVCH. This reconciliation determined that material (Alloy 690) procured to ASME B&PV Code, Section III 1989 Edition, no Addenda, Code Case 2142-1 and Code Case 2143-1 met all the requirements of ASME B&PV Code, Section III, 1998 Edition with Addenda through 2000. This removes the requirement to use of code cases imposed by the earlier Code year edition on the use of the Alloy 690 material, but maintains the qualification of the ANO-1 replacement RVCH.
- CALC-020638E101-04, “ANO-1 RV Closure Analysis W/Replacement Head” evaluated the RVCH closure joint region in conjunction with the use of the existing reactor vessel flange, closure studs, nuts, washers and internals. The analysis demonstrates that the replacement RVCH meets the design requirements of the design specification SPEC-ANO-M-587. The evaluation also demonstrates that the RVCH closure joint would meet the stress and fatigue requirements of ASME B&PV Code, Section III, 1989 Edition, Sub-section NB for Class 1 Components for the design life of 40 years. The maximum cumulative usage factor for the replacement RVCH closure region = 0.03 (< 1.0 maximum allowed by the design code). Additionally, the analysis determined the sealing ability of the closure joint for operational conditions would be equivalent to that of the existing configuration. Therefore the replacement RVCH has no adverse effect on the existing reactor vessel flange, closure studs, nuts washers and internals and meets all Code and design specification requirements.
- CALC-020638E101-09, “ANO-1 Segmented Skirt Weld Analysis Qualification”, analyzed the segmented skirt weld to the top of the replacement closure head. The weld is described on ANO-1 Dwg. M1B-400, “Specification Drawing for Replacement Reactor Vessel Closure Head ANO-1”. The result of this analysis qualifies the weld between the segmented skirt and the RVCH to the requirements of Subsection NB of the ASME Code Section III, 1989 Edition with no addenda. • CALC-020639E101-03, “ANO-1 SSS Support Skirt Analysis”, provides analytical confirmation that the ANO-1 Service Structure Support Skirt and the lower support Skirt Segments’ design meets the primary stress requirements of the ASME Code, Section III, 1989 Edition without addendum. This evaluation analytically confirmed that the bolted connections between the segmented flange and the support skirt lower flange, and the support skirt upper flange to the CRDSS lower flange meets the requirements of the ASME Code, Section III, 1989 Edition without addendum. Also this analysis qualified the support skirt and segmented skirt to the higher loads due to the increase in the service structure weight, which includes the added weight of the Service Structure Support Skirt.
- CALC-020638E101-06, “ANO-1 RVCH ASME Section III Code Reconciliation”, identified that the stress criteria are the same in both code editions with the following exceptions:

  In Table N-413 of the original Construction Code, the bending stress due to a through-wall thermal gradient is classified as a peak stress. In Table NB-3217 of the 1989 Edition of the ASME Section III, this stress, the equivalent linear stress, is classified as a secondary stress and, therefore, its effect must be included in the calculation of the range of primary-plus secondary stress intensity (NB-3222.2).

  The procedure for a detailed fatigue analysis in the 1989 Edition of the ASME Section III includes a provision for adjustment of the calculated alternating stress intensity by the ratio of the modulus of elasticity given on the design fatigue curve to the value of the modulus of elasticity used in the analysis (NB-3222.4(e)(4)). This requirement did not exist in the original Construction Code.

  CALC-020638E101-06 also identifies that the exceptions make the ASME Code Section III, 1989 Edition more restrictive than the original construction code and would therefore bound the original code.
- CALC-020638E101-08, “ANO-1 Appendix G RV Closure Head Limit”, updated the closure head heatup rate limits for the new closure head flange for normal operation conditions as well as for inservice leak and hydro-test conditions. These updated closure head limits are bounded by the current closure head limits used in the development of the ANO-1 Technical Specification Pressure/Temperature limits.
- CALC-020638E101-01, “ASME Design Report for Arkansas Nuclear One 1 Replacement Reactor Vessel Closure Head” provides professional certification that the replacement RVCH complies with the design requirements of the design specification SPEC-ANO-M-587 and the applicable criteria of the ASME B&PV Code, Section III, Division I, 1989 Edition with no Addenda.

Summary

The ANO-1 ASME Section XI Code of Record is the 1992 Edition, with portions of the 1993 Addenda. CALC-020638E101-02, "ANO-1 Replacement RV Closure Head Reconciliation", was prepared in accordance with the ASME Section XI Code of Record reconciling the replacement RVCH and CRDSS support skirt to the owner's original requirements and that any new requirements established in SPEC-ANO-M-587, "Design Specification for Reactor Vessel Closure Head Replacement ANO-1" have been met.

The quality requirements for the RVCH and support skirt do not change from quality requirements of the original components. The RVCH remains a safety-related seismically qualified. Additionally, the replacement RVCH's tests and exams will be performed in accordance with the applicable codes and specifications. The RVCH has been fabricated as designed and material traceability has been maintained.

Although the replacement RVCH will be fabricated from different materials, the materials have been evaluated and determined not to reduce the capability or functionality of any equipment important to safety. All materials are compatible with the environment in which they will be used. The material volume change is insignificant and would have no impact on any input or assumption used in the calculation of containment peak pressure for any FSAR identified accident.

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation <u>ONLY</u>? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.	☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?	☐ Yes ☒ No

BASIS:

Accidents and events evaluated in the FSAR that could be impacted by this proposed activity would be:

- Rod withdrawal at rated power
- Stuck-out, stuck-in, or dropped Control Rod Accident
- Rod Ejection Accident
- Loss of Coolant Accident
- Reactor Vessel Head Drop Over an Open Reactor Vessel

The pressure boundary capability of the replacement RVCH has been evaluated to be equivalent or better than the existing RVCH and that the functionality of the RVCH is maintained.

This activity will not change how Control Rod Drive Mechanisms are supported, restrained, powered, operated or provide control rod position indication; therefore there will be no change in occurrence to any accident associated with the control rods.

The impact of additional combined weight of the RVCH and Service Structure has been evaluated with the results determining that the additional weight was acceptable for the RCS supports and did not impact the reactor internals; and therefore would not increase the frequency of an accident. A new maximum head lift height above the reactor vessel was established. This new maximum height still provides margin to clear the top of the Alignment Guide Studs while staying within the risk associated with carrying heavy loads over the open reactor vessel during refueling (such as the reactor vessel head). "Unit 1 Reactor Vessel Closure Head Removal and Storage", procedure number 1504.007 will be revised to incorporate the new maximum lift height.

ER-ANO-2002-0639-000, “ANO-1 Service Structure Replacement”, qualified the replacement lifting configuration to lift and control the combined additional weight of the RVCH and Service Structure.

ER-ANO-2002-0640-000, “ANO-1 Polar Crane Up-rate to 190 Tons”, provides the qualification of the polar crane to lift and control the combined additional weight of the RVCH and Service Structure.

ER-ANO-2003-0954-000, “Downgrade to Augmented Quality Status of Thermal Insulation in the ANO-1 Reactor Building”, provided the evaluation for downgrading the RVCH insulation from “Q” to “Augmented Quality”. This ER determined that insulation classified as an augmented quality item would not result in a reduction in capability or change the function of the insulation. The RVCH insulation is not an initiator of any accident and changing the quality rating of the insulation will not cause it to become an initiator of an accident.

Therefore the replacement RV closure head would not introduce an increase in the frequency of occurrence of an accident previously evaluated in the FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

Types of malfunctions that are considered:

- Loss of coolant
- Loss of coolant flow (forced or natural circulation)
- Control rod position indication
- Erroneous parameter indications
- Loss of venting capability
- Brittle fracture
- Crack initiation and propagation
- Support failure
- Lack of heat removal
- Blockage of reactor building sump

The pressure boundary capability of the replacement RVCH has been evaluated to be equivalent or better than the existing RVCH and that the functionality of the RVCH is maintained. The replacement RVCH has no affect on the power supplied to, operational controls for or available indications for any SSC.

This activity will not change how Control Rod Drive Mechanisms are supported, restrained, powered, operated or provide control rod position indication; therefore there will be no change in occurrence to any malfunction associated with the control rods.

The impact of additional combined weight of the RVCH and Service Structure has been evaluated with the results determining that the additional weight was acceptable for the RCS supports and did not impact the reactor internals. The Reactor Vessel Head Stand and the floor loading in the head stand area were evaluated for the additional weight and were determined to be adequate to support additional weight and therefore would not increase the likelihood of a malfunction or compromise the function or mitigative capabilities of any SSC.

ER-ANO-2003-0954-000 provided the evaluation for downgrading the RVCH insulation from “Q” to “Augmented Quality”. This ER determined that insulation purchased as an augmented quality item would not result in a reduction in capability or change the function of the insulation. This impact of insulation used inside containment is one of the major regulatory issues with thermal insulation and the above ER determined that augmented quality insulation would have no affect on any issues inside the reactor building.

The insulation purchased to the augmented quality requirements is identical in form, fit and function to what would otherwise have been purchased as safety-related. Debris transport testing is equally applicable to both and has been performed on metal reflective insulation to confirm the continued operability of the ECCS pump suction. The primary differences between the purchase requirements for

"Q" and "Augmented Quality" insulation are; 1) 10CRF50 Part 21 requirements are not invoked, 2) testing of basic materials by the manufacturer is not required for materials not purchased from a supplier with a 10CRF 50 Appendix B quality assurance program (a certificate of conformance or certified materials test report continues to be required), 3) the qualification requirements of certain inspectors are relaxed (e.g., metal reflective insulation welding inspectors are required to be AWS certified but not 10CFR50 Appendix B qualified).

Therefore, this activity would not increase the likelihood of occurrence of a malfunction of any safety or non-safety SSC previously evaluated in the FSAR or creates any new malfunction.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

Accidents and events evaluated in the FSAR that could be impacted by this proposed activity would be:

- Rod withdrawal at rated power
- Stuck-out, stuck-in, or dropped Control Rod Accident
- Rod Ejection Accident
- Loss of Coolant Accident.
- Reactor Vessel Head Drop Over an Open Reactor Vessel

The replacement reactor vessel closure head has been evaluated and determined to meet or exceed all code requirements for fabrication, installation and testing of this classification of component. It has been demonstrated and documented that all the components meet the code and owner's requirements as required by ASME B&PV Code, Section XI. No new radiological sources, processes or flow paths have been introduced. The new materials have been evaluated to be equivalent or better than the material they will replace.

The impact of additional combined weight of the RVCH and Service Structure has been evaluated with the results determining that the additional weight was acceptable for the RCS supports and did not impact the reactor internals; and therefore would not increase the consequences of an accident.

CALC-020638E101-16, "ANO-1 RV Head Lift Height" provided a new maximum head lift height above the reactor vessel. This new maximum height remains within the risk associated with carrying heavy loads over the open reactor vessel during refueling (such as the reactor vessel head) and providing the basis for expecting the consequences of dropping the reactor vessel head to remain unchanged.

This activity will not significantly increase the radiological activity levels in any contaminated system or reduce the capability of any barrier limiting dose rates to onsite or offsite personnel. This activity will not change any actuation setpoint, reduce any margin to existing setpoints, reduce the capability of any SSC's accident mitigating functions or change how any function is performed. Therefore, there would be no increase in the consequences of an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

Types of malfunctions that are considered:

- Loss of coolant
- Loss of coolant flow (forced or natural circulation)
- Control rod position indication
- Erroneous parameter indications
- Loss of venting capability
- Brittle fracture
- Crack initiation and propagation
- Support failure
- Lack of heat removal
- Blockage of the reactor building sump

The replacement RVCH and all associated appurtenances have been verified to meet the code and owner's requirements as required by ASME B&PV Code, Section XI.

ER-ANO-2003-0954-000 qualified the augmented quality insulation for use in the reactor building and determined that the new insulation would have no adverse impact on the reactor building sump or the functionality of any SSC inside the reactor building.

The impact of additional combined weight of the RVCH and Service Structure has been evaluated with the results determining that the additional weight was acceptable for the RCS supports and did not impact the reactor internals.

No new radiological sources, processes or flow paths have been introduced. The new materials have been evaluated to be equivalent or better than the material they will replace. This activity will not significantly increase the radiological activity levels in any contaminated system or reduce the capability of any barrier limiting dose rates to onsite or offsite personnel. This activity will not change any actuation setpoint, reduce any margin to existing setpoints, reduce the capability of any SSC's accident mitigating functions or change how any function is performed. Therefore this activity would not increase the consequences of a malfunction of any safety or non-safety SSC previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The replacement components have been verified to meet the code and owner's requirements as required by ASME B&PV Code, Section XI. Maintaining strict adherence to the applicable code requirements and owner's specifications provides assurance that the capability and functionality of the replacement reactor vessel closure head will meet or exceed the original closure head's abilities.

ER-ANO-2003-0954-000 qualified the augmented quality insulation for use in the reactor building and determined that the new insulation would have no adverse impact on any SSC inside the reactor building.

The additional weight of the RVCH and Service Structure was evaluated to be acceptable for the RCS supports and did not impact the reactor internals.

This activity will not require a change to how any SSC is controlled or operated.

Therefore this activity would not create the possibility for an accident of a different type than previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

All components have been verified to meet the code and owner's requirements as required by ASME B&PV Code, Section XI. Maintaining strict adherence to the applicable code requirements and owner's specifications provides assurance that the capability and functionality of the replacement reactor vessel closure head will meet or exceed the original closure head's abilities.

ER-ANO-2003-0954-000 qualified the augmented quality insulation for use in the reactor building and determined that the new insulation would have no impact on the functionality or capability of any SSC inside the reactor building.

The additional weight of the RVCH and Service Structure was evaluated to be acceptable for the RCS supports and did not impact the reactor internals.

This activity will not require a change to the operation of the plant nor will it change how any SSC is controlled or operated. No setpoint will be changed nor will the operation change of any equipment required to mitigate the consequences of an accident.

The above evaluations ensure that a malfunction of a structure, system or component important to safety would remain bounded by existing analyses and different results than previously evaluated in the FSAR will not occur.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?

☐ Yes
☒ No

BASIS:

The review of the DBLFPBs identified the following could be possibly impacted by the reclassification of the insulation or the additional combined weight of the RVCH and Service Structure:

- DNBR/MCPR: Neither the insulation reclassification nor the change in component weight would have an impact on the RCS volume, flow rates, flow paths or primary flow capabilities therefore all events having MDNBR acceptance criterion will still meet that acceptance criterion.
- Primary Pressure: This activity will not change any primary system parameter or limit. There would be no reduction in margins to any limit and there would be no change to how any SSC is operated or controlled. The transient and accident mitigating capabilities of all SSC's will be maintained; therefore all events having a maximum primary pressure acceptance criterion will still meet that acceptance criterion.
- RCS Boundary Stresses: The replacement reactor vessel closure head has been evaluated and determined that the stresses associated with the RCS boundary were acceptable for normal, upset, or faulted conditions. All components have been verified to meet the code and owner's requirements as required by ASME B&PV Code, Section XI. The additional weight of the RVCH and Service Structure was evaluated to be acceptable for the RCS supports; therefore the RCS boundary stresses remain acceptable.
- Containment Design Pressure: The reclassification of the replacement RVCH insulation does not decrease the insulating capability of the replacement insulation panels. The replacement insulation has better insulating capabilities which will reduce the heat losses from the primary system to the containment atmosphere. This would assist in preventing the peak containment temperature established for ensuring the integrity of the containment from being exceeded.

Therefore no design basis limit for a fission product barrier as described in the FSAR will be exceeded or altered by the installation of the replacement RVCH or CRDSS support skirt.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☒ No

BASIS:

The safety/quality classification of insulation is not an element of any evaluation methods described in the FSAR. The impact of the combined additional weight of the RVCH and Service Structure has been evaluated and determined to be acceptable: therefore the inputs used for the design bases and the safety analyses remain valid. Neither of the above issues would impact methods of evaluation described in the FSAR used in establishing the design bases or the safety analyses.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-016

I. OVERVIEW / SIGNATURES

Facility: **ANO - Unit 1**

Document Reviewed: ER-ANO-2002-1078-015, ANO-1 SG/RVCH Replacement - OTSG Removal & ROTSG Preparation/Installation

Change/Rev.: 0 (CN-1)

System Designator(s)/Description: Steam Generators (E24A & E24B)

Description of Proposed Change:

ER-ANO-2002-1078-015, ANO-1 SG/RVCH Replacement - OTSG Removal & ROTSG Preparation/Installation provides for the pre-outage preparation of the ROTSGs (E24A and E24B) and the cutting of the Reactor Coolant System (RCS) hot and cold leg piping and connecting branch line piping/tubing to allow for removal of the Original Once Through Steam Generators (OOTSGs). Once the Replacement Once Through Steam Generators (ROTSGs) are positioned, this modification package provides for the reconnection of OTSG support and restraint systems and installation of a new hot leg piping section and re-welding of the hot and cold leg piping.

Changes and activities addressed by this review include:

- A segment of the hot leg piping (CCA-1-36") leading to each steam generator will be replaced with a new hot leg elbow section and a new High Point Vent Nozzle and Hot Leg Level Monitoring Nozzle
- The hot leg piping (CCA-1-36") and cold leg piping (CCA-1-28") at the OTSG will be required to be cut and reattached by welding to the ROTSG.
- Upper Lateral Restraints (ULR) will be disconnected or cut and restored
- OTSG Foundation and Base Support removal and reinstallation activities
- RCS pipe (rupture) restraints (MK-33 and MK-34) will be abandoned in place and made inactive supports.
- RCS High Point Vent and Nitrogen Supply piping (CCA, CCB, HSC, and HCD Piping Class) will be temporarily removed and restored
- Temporary removal and reinstallation of the RCS Hot Leg Level monitoring piping (CCA-1-1") and tubing (LT-1190 & LT-1195).
- The RCS Temporary Level tubing will be removed and reinstalled.
- Insulation will be removed from the hot and cold leg piping as required for machining, welding and temporary support activities. Existing RCS piping metal reflective insulation (MRI) that is removed will be reinstalled under ER-ANO-2002-1078-015. Installation of new MRI will be installed by ER-ANO-2002-1078-016, ANO-1 SG/RVCH Replacement -Insulation.
- The attachment of temporary restraints to the hot leg and cold leg piping in order to prevent pipe deflection after the RCS piping cuts are made. These temporary restraints will be removed following installation of the ROTSG and with the establishment of sufficient welding of the RCS piping when the restraints are no longer needed.
- Welding of lifting trunnions onto the OOTSGs to allow for OTSG removal by ER-ANO-2002-1078-011, ANO-1 SG/RVCH Replacement – Rigging & Handling – Inside Containment and removal of the lifting trunnions from the ROTSGs.
- Pipe end decontamination activities involving the open ends of the RCS hot and cold leg piping after severence
- Removal of photogrammetry targets installed during 1R17 after templating operations are complete.
- Provision of insulation support and grounding strap mounting holes on the ROTSG
- Modification of Surge Line supports

ER-ANO-2002-1078-015 performs a number of preparatory activities to the OOTSGs prior to their removal from the Reactor Building such as installation of shield plugs and application of a contamination encapsulant. Since these activities are performed on components (the OOTSGs) that have been permanently removed from service, they are not addressed further in this review.

The OOTSG/ROTSG differences and qualification of the ROTSGs as equivalent replacement SG components are addressed in ER-ANO-2002-1381-000, ANO-1 SG/RVCH Replacement - ROTSG Qualification Package. Removal and reconnection of other connected piping and instrumentation is addressed in other modification packages.

All the changes addressed in this review have been analyzed and evaluated in ER-ANO-2002-1078-015. The functionality and capability of all structures, systems and components at ANO-1 are maintained or enhanced. All FSAR described design functions will continue to be available without any degraded capability. ANO-1 will continue to operate within the requirements of the current Operating License (OL) and Technical Specifications (TS), therefore this activity may be performed without prior Nuclear Regulatory Commission review and approval.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☒	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-016)**	**Sections I, II, and IV required**

Preparer: John Pearman / **ORIGINAL SIGNED BY JOHN PEARMAN** / NAGI / SG-RVCH / 4-12-05
Name (print) / Signature / Company / Department / Date

Reviewer: Kirk Dixon / **ORIGINAL SIGNED BY KIRK DIXON** / ADDECO / SG-RVCH / 4-12-05
Name (print) / Signature / Company / Department / Date

OSRC: J.R. Eichenberger / **ORIGINAL SIGNED BY J.R. EICHENBERGER** / 4-14-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	Sections 4.1.2.3, 4.1.2.5.2, 4.2.1.2, 4.2.2.4, 4.2.6.3, 4.2.6.6, , 4.3.2, 4.3.12.3, Tables 4-6, 4-9 and 4-12, Section 16.3.9
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ **Yes** ☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. **Basis**

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

ER-ANO-2002-1078-015 provides for severing of the RCS piping for steam generator removal and for replacement of a 135-degree hot leg elbow.

Operating License/Technical Specifications (TS)

The Steam Generators are discussed in TS Sections 3.4.4, 3.4.5, 3.4.6, 3.4.7, and 5.5.9. TSs 3.4.4, 3.4.5, and 3.4.6 specify the required availability of the reactor coolant loops in Modes 1 through 5. Since the RCS will be restored to its licensed condition following installation of the ROTSGs, no changes to these TSs are needed. A License Amendment was submitted on September 30, 2004 by EOI to revise the S/G surveillance requirements in Technical Specification 5.5.9. The hot leg elbows, RCS piping welds, Surge Line supports, Upper Lateral Restraint steel and bearing pads, and attachment bolting material, are not addressed in the Technical Specifications.

FSAR

The steam generators are discussed in Sections 4.2.2, 4.3.4, and Table 4-4. The steam generators are also mentioned in numerous other sections in the FSAR. Updates to steam generator discussions and descriptions will be made by ER-ANO-2002-1078-000, ROTSG Qualification Package. Any updates to the FSAR required as part of Code qualification of the RCS system including RCS piping stress analysis will be made by ER-ANO-2002-1381-000, except as noted.

The RCS piping and materials, including the hot leg elbows, is discussed in Sections 4.1.3.2, 4.2.2.4, 4.3.2 and Tables 1-1, 4-2, 4-6, 4-9 and 4-21a. Revisions are needed to Section 4.2.2.4 and Tables 4-6 and 4-9.

The RCS piping welds are discussed in Sections 4.3.2, 4.3.12.3, and Table 4-12. Section 4.3.12.3 will be revised to reflect the performance of magnetic particle and liquid penetrant tests prior to the inservice leak test of the new RCS piping welds and Table 4-12 will be revised to correct a typographical error for field acceptance of RCS piping weldments from UT to RT . This is addressed in the 50.59 Evaluation portion of this Review.

Upper lateral restraints are discussed in Sections 4.2.1.2, 4.2.6.3 and 4.2.6.6. Sections 4.2.1.2 and 4.2.6.6 will be revised to reflect the use of leak-before-break. This is addressed in the 50.59 Evaluation portion of this Review.

The steam generator foundations are discussed in Sections 4.2.6.3 and 5.3.1. Section 4.2.6.3 will be revised to reflect the use of leak-before-break. This is addressed in the 50.59 Evaluation portion of this Review.

The RCS pipe restraints are discussed in Sections 4.2.6.6, 4.2.7 and 16.3.9. Section 4.2.6.6 and 16.3.9 will be revised to reflect that these pipe restraints have been abandoned in consideration of use of Leak-Before-Break. This is addressed in the 50.59 Evaluation portion of this Review.

Reaction and loss of coolant loads are discussed in Sections 4.1.2.3 and 4.1.2.5.2. Sections 4.1.2.3 and 4.1.2.5.2 will be revised to reflect the use of leak-before-break for the upper lateral restraints and lower base support. This is addressed in the 50.59 Evaluation portion of this Review.

The RCS Surge line is discussed in Section 4.1.2.2, 4.2.2, 4.2.4.6, 4.2.6.2, 4.4.1 and Tables 1-2, 4-5, 4-6, 4-9, 4-29a. No changes to these FSAR sections or any FSAR figures are required with regard to the RCS surge line and supports. The following activities, due to potential to reduce the level of conservatism used in SSC evaluation or changes required in the FSAR as noted above, are being reviewed in Section IV – 50.59 Evaluation.

- Replacement hot leg elbow and High Point Vent Nozzle and Hot Leg Level Monitoring Nozzles – The 135° section of hot leg piping will be replaced with a new equivalent elbow in order to provide welding personnel a low-dose area for welding the elbow to the hot leg riser. The new elbow will be a near identical replacement in that the elbow will be fabricated from the same SA-516 Grade 70 rolled plate as the current elbows. Other material changes were necessary and desirable. The straight section of the elbow will be fabricated from the same SA-516 Grade 70 material in lieu of A-106 Grade C seamless pipe that is currently used in the elbow. The Inconel (Alloy 600) high point vent nozzle will be changed to an Alloy 690 nozzle. The RCS hot leg level monitoring nozzle that was previously field installed will be installed by the elbow fabricator using Alloy 690 nozzles. The new hot leg elbows will be fabricated to the ASME Code, Section III 1998 Edition with 2000 Addenda. The new hot leg elbows are considered equivalent to the existing elbows and therefore do not increase the frequency of a LOCA.

- RCS Pipe Cutting and Re-welding – The hot and cold leg pipe cuts will be made with a rotating lathe and cutting tools that will minimize the introduction of debris into the RCS piping. The RCS piping will be severed in the cold leg piping in a 22.5° inclined plane cut through the 45° cold leg elbow adjacent to the OOTSG. The hot leg piping will be severed at the OOTSG inlet nozzle and at a second location at approximately 135° from the inlet nozzle in the hot leg riser pipe. Following severance of other attached piping and instrumentation by other ERs, the OOTSGs will be removed by ER-ANO-2002-1078-011. The cold leg pipe end, hot leg riser pipe and both ends of the new hot leg elbows will be templated and then machined to provide a close tolerance fit with the ROTSG nozzles. In order to achieve final fit-up of the replacement hot leg elbow with the premachined ROTSG hot leg nozzle, jacking of the RCS hot leg riser may be used as a contingency. An automated, narrow groove, two-sided GTAW weld process will be used to attach the ROTSGs to the cold legs, hot legs elbows and hot leg riser. The interior surface of the cold leg and hot leg welds will be clad with stainless steel to provide corrosion resistance to the reactor coolant. The ROTSGs are supplied at delivery with an extended pipe section and attached elbow at each cold leg nozzle fabricated from ASME SA 508 Class 3A material. The equivalence of this change in material is addressed in ER-ANO-2002-1381-000. The four new RCS field welds per steam generator will be examined by radiography, magnetic particle, and dye penetrant in accordance with the ASME Code. The completed RCS welds will also be inspected in accordance with Section XI of the ASME Code. The new RCS pipe welds are considered equivalent to the existing RCS welds and do not increase the possibility of a loss of coolant accident (LOCA).

- Upper Lateral Restraint (ULR) - ER-ANO-2002-1078-015 provides for the temporary removal of the steam generator ULR to allow for removal of the OOTSGs and the installation of the ROTSGs. With a few minor changes, the ULR will be returned to a configuration equivalent to their original configuration. The upper lateral restraint steel and bearing pads will be removed and later reinstalled to the design configuration using an equivalent bolting material. Where there are welded connections in places where steel members were cut, the members will be joined together utilizing full penetration welds. The existing bolts larger than 1 ½” are no longer available in ASTM A490 and these bolts will be replaced with equivalent bolts meeting the requirements of ASTM A 354, Grade BD. The Meehanite bearing pads will be installed to the evaluated design condition. The ULR structure is considered equivalent to the existing ULR structure and does not increase the possibility of a loss of coolant accident (LOCA).

- OTSG Foundations and Base Support - The concrete curb and adjacent grout surrounding the steam generator base support skirt will be removed to the extent necessary to allow removal of the OOTSGs and to permit machining and/or welding activities. The lower base support will be unbolted from the OTSG to allow removal. The ROTSGs will be anchored to the existing imbedded anchor bolts with nuts and washers.

- <u>RCS pipe (LOCA) restraints</u> - Per ANO FSAR Paragraph 4.2.6.6, the RCS pipe (28" and larger) whip restraints are no longer required and may be removed as needed to facilitate maintenance or other activities. Therefore, it is permissible to abandon the MK-33 and MK-34 LOCA restraints in place.

ER-ANO-2002-1078-015 takes credit for leak-before-break in the analysis of the ULRs, lower base support and pipe restraint abandonment as described above. On December 12, 1985 (letter from D. M. Crutchfield, NRC, to L. C. Oates, B&W OG), the NRC approved the B&W Owners Group Topical Report BAW 1847, Rev.1, "Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W NSS", September 1985. ANO correspondence 1CNA028604 and 1CAN108607 address application of BAW-1847and BAW-1889 for leak-before-break. This approval was previously used to reduce the LOCA loadings on fuel assemblies (FSAR Section 3.3.3.3.2.1), to justify pipe whip restraints no longer being required on RCS piping (FSAR Section 4.2.6.6), and for the removal of the wire tie ropes on the reactor coolant pumps (FSAR Section 4.1.2.3).

The remainder of the changes or activities performed by ER-ANO-2002-1078-015 and their impact to the FSAR are being addressed in Section III – 50.59 Evaluation Exemption portion of this Review.

- RCS high point vent and nitrogen supply piping
- Temporary removal and reinstallation of the RCS Hot Leg Level monitoring piping and tubing (LT-1190 & LT-1195)
- The RCS Temporary Level tubing will be removed and reinstalled.
- Insulation will be removed from the hot and cold leg piping as required for machining, welding and temporary support activities. Existing RCS piping MRI that is removed will be reinstalled under this ER. Installation of new reflective metal insulation will be installed by ER-ANO-2002-1078-016, ANO-1 SG/RVCH Replacement -Insulation.
- Temporary restraints will be attached to the hot leg and cold leg piping in order to prevent pipe deflection after the RCS piping cuts are made. These temporary restraints will be removed following installation of the ROTSG and with the establishment of sufficient welding of the RCS piping to the point that the restraints are no longer needed.
- Welding of lifting trunnions onto the OOTSGs to allow for OTSG removal by ER-ANO-2002-1078-011, ANO-1 SG/RVCH Replacement – Rigging & Handling – Inside Containment
- Pipe end decontamination activities
- Removal of photogrammetry targets installed during 1R17

<u>Test or Experiment Consideration</u>

The cutting out of the OOTSGs and re-welding of the ROTSGs and new hot leg elbows will not result in the RCS being utilized or controlled in a manner that is outside of the reference bounds of the design as described in the FSAR. Once the ROTSGs are welded into the RCS and other connecting piping and instrumentation are restored by other ERs, the RCS will undergo an inservice leakage test in accordance with ASME Code Case N-416-2 where applicable. The inservice leakage test is not considered a test or experiment for 50.59 Evaluation and will not be conducted in a mode which has not been previously analyzed.

The replacement of the highpoint vent nozzle with Alloy 690 material maintains the corrosion-resistant boundary of the RCS as described in FSAR Section 4.3.2. AREVA Document 32-5017980-00, ANO-1 RCS Structural Evaluation found the stresses at the high point vent nozzle to be acceptable. Replacement of the high point nozzle with an improved corrosion-resistant material is not considered a test or experiment.

The replacement of the upper lateral bolts with an available, equivalent material is not considered a test or experiment.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LRS 50.59 – Unit 1 - Autonomy	Steam generator*, “OTSG”, “reactor coolant system”, “hot leg”, “cold leg”, trunnion*, foundation*, RCS weld*, “ULR”, “restraint”, “lateral”, “meehanite”, “LOCA”, “leak before break”, “MK-33”, “reactor coolant piping”, “anchor bolt*”, “steam generator support”, “ASTM A490” , “A-106”, LBB, surge line, surge pip*
LBDs/Documents reviewed manually:	
ANO Unit 1 FSAR	Chapter 4, Sections 5.1, 14.2.2.5 and 16.3.9; Tables 4-2, 4-6, 4-9, 4-12, 4-21a and 14-49; Tables 1-2, 4-5, 4-6, 4-9 and 4-29a, Figure 4-6
ANO Unit 1 Facility Operating License, Technical Specifications, COLR, and Technical Requirements Manual	
Entergy Quality Assurance Program Manual, Revision 11, April 30, 2004	

5. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☐ **Yes**
☒ **No**

If “YES,” list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☒ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: ____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10 CFR 50.59 Review Program Guidelines for guidance.

ER-ANO-2002-1078-015 provides for work activities that are needed to sever the reactor coolant piping to allow for removal of the OOTSG. Following their placement, the ROTSGs will be welded to the machined reactor coolant piping. The unit will be in the defueled condition during the cutting and re-welding of the RCS piping.

This exemption addresses the following attributes:

RCS High Point Vent and Nitrogen Supply Piping - To provide sufficient clearance for the removal of the OOTSGs and Hot Leg elbow section, it is necessary to temporarily remove a section of the RCS High Point Vent and Nitrogen Supply piping. To disconnect the RCS High Point Vent and Nitrogen Supply piping (CCA, CCB, HSC, and HCD Piping Class), cuts will be made on the piping. Cuts will be at existing weld interfaces or, if needed, cuts will be made and a welded coupling added as provided for by design. The RCS High Point Vent and Nitrogen Supply piping will be reinstalled to its design configuration and reconnected to the replacement High Point Vent nozzle. All pinned constant supports will be unpinned after the completion of welding the RCS High Point Vent piping. This activity does not affect the design function, method of controlling a design function or method of evaluation as described in the FSAR for the RCS High Point Vent and Nitrogen Supply piping.

RCS Temporary Level Tubing - Connected to the Nitrogen Supply piping valve N2-134 in the north cavity is tubing for RCS Temp Level. To prevent this tubing from being damaged during the removal activities, the tubing will be temporarily removed by disconnecting at existing compression fittings. This tubing will be restored to its design condition. This activity does not affect the design function, method of controlling a design function or method of evaluation as described in the FSAR.

Insulation – Prior to the installation of RCS Temporary Restraints and the severance of the OOTSGs from the RCS, insulation must be removed from the RCS Hot and Cold Leg piping. Currently the sections of the RCS piping are insulated with either Metal Reflective Insulation (MRI) or blanket type insulation. Blanket type insulation is installed on the RCS Hot Leg from the OOTSG Hot Leg nozzle through the 180° elbow and installed on the RCS Cold Legs from the OOTSG Cold Leg nozzle to approximately 27 inches from the outside surface of the OOTSG support skirt. Insulation at these areas must be removed to accommodate cutting, machining and welding equipment. Once removed, blanket type insulation will be discarded. After the completion of the RCS welding and acceptance of Non Destructive Examination (NDE), the RCS piping will be reinsulated. MRI removed from the RCS Hot and Cold Legs will be reinstalled to its original configuration. The blanket type insulation previously removed from the RCS Hot and Cold Leg piping as part of ER-ANO-2002-1078-015, will be replaced with MRI by ER-ANO-2002-1078-016. This activity does not affect the design function, method of controlling a design function or method of evaluation as described in the FSAR.

Temporary Restraints - The replacement of the OTSGs requires their severance from RCS Hot and Cold Leg piping. Temporary restraints for the RCS piping are needed to provide deadweight support of the free ends and to minimize pipe deflection due to pre-existing cold spring forces and moments that may exist from welding and fit-up during the original installation of the RCS. Maintaining the piping position by temporary restraints will insure proper fit-up with the ROTSG RCS nozzles. Temporary supports on the RCS will be removed after sufficient welding has been achieved on each respective RCS pipe leg. This activity does not affect the design function, method of controlling a design function or method of evaluation as described in the FSAR.

Lifting Trunnions – Trunnions are welded onto the original OTSGs as part of ER-ANO-2002-1078-015 in support of rigging and handling activities performed under ER-ANO-2002-1078-011. The welding of trunnions is performed on the original OTSGs after they are no longer required for decay heat removal and are no longer required to be operable components. Therefore, these activities do not have any impact on and do not affect the design function, method of controlling a design function or method of evaluation as described in the FSAR. Temporary lifting trunnions are provided on the ROTSGs. They will be removed from the ROTSGs prior to entry into Mode 4 and will not affect the operability of the ROTSGs.

Pipe End Decontamination (PED) Process – The ends of the RCS piping will be decontaminated using a mechanical decontamination process. This process is based on a two-step abrasive blasting technique using electro-corundum to remove the radioactive oxide layer followed by a glass bead treatment to polish the surface. There is virtually no pipe wall (cladding) metal removal due to the decontamination process. The pipe pressure rating, structural capacity of the piping, and corrosion resistance are not affected. This activity does not affect the design function, method of controlling a design function or method of evaluation as described in the FSAR.

Photogrammetry Target Removal – Twenty (20) targets composed of Aluminum were installed during 1R17 on the cavity walls in support of photogrammetry operations performed during 1R17 and to be performed in 1R19. These targets had minimal affect on calculations that tracked aluminum surface area and net free reactor building volume and are bounded by values reported in the FSAR. Updates to the appropriate calculations are made by ER-ANO-2002-1078-015 to reflect the removal of the aluminum from the reactor building. This activity does not affect the design function, method of controlling a design function or method of evaluation as described in the FSAR.

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.

☐ Yes
☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The accident previously evaluated in the FSAR that is potentially impacted by this ER is the Loss of Coolant (LOCA) which is described and evaluated in FSAR Section 14.2.2.5. The activities to be performed by ER-ANO-2002-1078-015 that potentially impact the integrity of the RCS pressure boundary include:

The mechanical properties of the welds were evaluated in ANO Calculation 021381E101-212, "ANO Unit 1 EOTSG Replacement Piping Materials Evaluation for LBB", Rev. 0 to demonstrate that the leak-before-break design basis conclusions of the B&W Owners Group Topical Report BAW 1847, Rev. 1 remain valid for the material and weld changes of the steam generator replacement project. This report concluded that the topical report remained bounding.

The new hot leg elbows were included in the ANO Calculation 021381E101-212, Rev. 0 that concluded that the BAW-1847, Rev. 1 Topical Report for the leak-before-break design basis remained bounding. Refer to ERANO-2002-1381-000 and 50.59 Reviews performed for this engineering response for further evaluation of the ROTSGs and supplied cold leg piping with attachment welds to the ROTSG with regard to leak-before-break. ER-ANO-2002-1078-015 does not perform cutting or welding on secondary side piping. Therefore, secondary side accidents are not impacted by this ER.

It is therefore concluded that application of the currently approved LBB methodology will maintain the current licensing basis of the RCS pressure boundary and will not increase the frequency of any accident previously evaluated in the FSAR..

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

ER-ANO-2002-1078-015 provides for the replacement of the hot leg elbows and the addition of four new narrow groove welds per loop. These changes were addressed in ANO Calculation 021381E101-212, Rev. 0 and were found to be acceptable. Other material changes are needed due to the unavailability of identical materials.

The ULRs will be temporarily removed and later re-installed to provide clearance for removal of the OOTSGs and installation of the ROTSGs. The ULR structural bolts larger than 1½ inch diameter will be replaced with new bolts meeting the requirements of ASTM A354, Grade BD. The previous ASTM A-490 bolts are no longer available. The new bolts are considered equivalent material. At intersecting plate section that are reattached by full penetration welds access holes have been created and will not be restored in the ULR structural steel. As discussed in FSAR Section 4.2.2.2 and 4.2.6.3, the ULR was designed to resist lateral loads from a 36 inch RCS line break. Elimination of this break was justified in BAW 1847, Rev 1 and approved for the B&W Owners Group plants by the NRC on December 12, 1985. This is addressed in ANO CALC-021381E101-68, "ANO-1 RCS Structural Evaluation" (AREVA Document 32-5017980-00). Evaluation of the loads on the ULRs by ANO CALC #23 and the loads on the ULRs have been found acceptable.

The cylindrical skirt of the OOTSG is anchored to the Reactor Building floor with forty-eight 2½ inch diameter bolts, washers, hex nuts, and jam nuts and are reassembled to the design requirements. The ROTSGs will utilize the existing forty-eight imbedded anchor bolts. The holes in the support skirt of the ROTSG are fabricated with a larger diameter to facilitate fit-up of the ROTSG to the anchor bolts. As discussed in FSAR Section 4.2.6.3, the steam generator foundation is designed to restrain the steam generator under the combined forces of a cold leg break and a simultaneous MHE. Analysis of the base support loads in ANO CALC-021381E101-68, "ANO-1 RCS Structural Evaluation" (AREVA Document 32-5017980-00), took credit for leak-before-break (BAW 1847, Rev. 1) and showed that the loads from the ROTSG analysis were less than the loads generated in the previous analysis, and are therefore acceptable.

It is therefore concluded that the changes to the RCS piping (new welds and hot leg elbow) and steam generator supports (ULR and base support) meet the licensing basis criteria as documented in the FSAR, are of equivalent reliability, and therefore do not increase the likelihood of occurrence of a malfunction of a structure, system, or component previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

FSAR Section 14.2.2.5 evaluates a spectrum of pipe breaks in the RCS. A break is assumed to occur instantaneously and may be located anywhere in the RCS pressure boundary. Breaks may be double-ended with unobstructed flow from both ends or longitudinal splits. ER-ANO-2002-1078-015 potentially impacts the LOCA analysis by adding four additional RCS welds per loop and by replacing a 135° section of the hot leg elbow at the inlet to each steam generator. The FSAR reports the results of a LOCA where all the fuel rod gap activity is released to the RCS. Since a break in the cold leg portion will cause the worst results due to more core uncovering, a hot leg break would not be limiting. The FSAR also reports the results of an MHA (Maximum Hypothetical Accident) which is non-mechanistic and bounds the LOCA dose consequences event as just discussed. Since the current analyses make conservative assumptions for source term activity based on worse events than could occur with these activities, none of the activities planned by ER-ANO-2002-1078-015 will increase the dose consequences from a LOCA.

Other accidents evaluated in the FSAR (Steam Line Failure, Steam Generator Tube Failure, and Rod Ejection Accident) have calculated off site dose consequences. These accidents implicitly take credit for an intact RCS piping system in mitigating the consequences of the accident and cooling the core. Since the hot leg and cold leg piping will be returned to an equivalent design condition, this assumption remains valid.

The changes to the RCS piping were demonstrated to be equivalent to the current piping as discussed in response to Question 1. Therefore, the consequences of accidents currently evaluated in the FSAR will not be increased.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The components impacted by ER–ANO-2002-1078-015 include the RCS piping, ULRs, and steam generator lower base support.

As discussed in response to Questions 1 and 2, the new RCS welds and new hot leg elbows are considered equivalent to the current components. No increased reliance is placed on these components to perform their safety function as a part of the RCS pressure boundary.

As discussed in response to Question 2, the ULS and SG base supports will be returned to an equivalent design configuration.

As documented in ANO CALC-021381E101-68, "ANO-1 RCS Structural Evaluation" (AREVA Document 32-5017980-00), lesser loads have been placed on the ULR and SG base supports due to the inclusion of LBB in the evaluation of pipe break loads.

Therefore, the planned changes to the RCS piping and SG ULRs and base supports will not increase the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

ER-ANO-2002-1078-015 provides for severing the RCS hot leg and cold leg pipe, replacement of a 135° section of hot leg piping and other activities such as removal of the ULRs and removal of the SG lower base support fasteners. These activities are needed to remove the OOTSGs. The ROTSGs are then welded to the RCS piping in accordance with ASME Code Section XI, IWA-4000, as required by FSAR Section 4.1.3.2. At the completion of the ROTSG activities, the RCS will have been returned to an equivalent condition. The failure of the replacement welds and hot leg elbows are bounded by the current LOCA analysis in FSAR Section 14.2.2.5.

Therefore, ER–ANO-2002-1078-015 does not create the possibility of an accident of a different type than any previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

As discussed in Chapter 4 of the FSAR, the Reactor Coolant System is designed to protect against a number of adverse conditions and failure modes including:

1- Over pressure
2- Mechanical and thermal stresses
3- Cyclical loads
4- Seismic loads
5- Loss of coolant loads
6- Chemical corrosion
7- Neutron irradiation

The activities performed by ER-ANO-2002-1078-015 that potentially impact these failure modes include the additional four RCS welds per loop, replacement of a 135° section of each hot leg elbow, and removal and reinstallation of the SG supports (ULR and base support).

The four new welds per loop will be performed per the ASME Code Section XI, IWA 4000. This assures that the new welds are qualified for the loads listed above. The inside surface of the weld will be clad with stainless steel to resist corrosion. The new welds are at a sufficient distance from the reactor core that neutron exposure is not a concern.

As discussed in response to Question 1, the new hot leg elbows will be fabricated and inspected in accordance with Section III of the ASME Code. The elbow sections will be internally clad with stainless steel to resist corrosion.

The ULR and lower base supports will be returned to their design configuration including equivalent bolting/fastener materials as design specified. Evaluations were performed to demonstrate that the RCS pressure boundary design basis is maintained and are documented in ER-ANO-2002-1381-000.

Therefore the activities performed by ER-ANO-2002-1078-015 will not create the possibility for a malfunction of a structure, system, or component to safety with a different result.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?

☐ Yes
☒ No

BASIS:

Activities performed by ER-ANO-2002-1078-015 will have no impact on the design basis limits for the fuel cladding or containment pressure.

ER-ANO-2002-1078-015 does have the potential to impact the RCS boundary stress limits. In order to achieve fit-up of the new hot leg elbow and the hot leg pipe, it may be necessary to apply a jacking load on the hot leg riser pipe. CALC-ANO-ER-05-013, Rev. 0 permits the application of a jacking load of up to 190 kips in a single direction or 130 kips in two simultaneous directions on the hot leg piping.

The ROTSGs will be re-welded into the RCS pressure boundary with the addition of four new narrow groove welds per loop. The final weld profile has the potential for creating a high stress area. However, the fit-up and alignment requirements of ASME Section III Subsection NB will be followed thereby limiting the stresses introduced into the RCS pressure boundary by these new welds.

The mechanical properties of the welds and the new hot leg elbows were evaluated in ANO Calculation 021381E101-212, "ANO Unit 1 EOTSG Replacement Piping Materials Evaluation for LBB", Rev. 0 to demonstrate that the leak-before-break design basis conclusions of the B&W Owners Group Topical Report BAW 1847, Rev. 1 remain valid for the material and weld changes of the steam generator replacement project.

The RCS pipe machining, welding and NDE will be performed to approved procedures and will satisfy the requirements of ASME Sections III and XI. The pre-service leak test will be performed of the modified piping sections to ensure the weld joints and welded connections do not exhibit leakage at normal service conditions.

Therefore, it is concluded that the activities performed by ER-ANO-2002-1078-015 will not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

The NRC's approval of leak-before-break was previously used to reduce the LOCA loadings on fuel assemblies (FSAR Section 3.3.3.3.2.1), to justify pipe whip restraints no longer being required on RCS piping (FSAR Section 4.2.6.6), and for the removal of the wire tie ropes on the reactor coolant pumps (FSAR Section 4.1.2.3). The basis for the NRC approval was the B&W Owners Group Topical Report BAW-1847, Rev. 1. The new materials and welds that are introduced into the RCS by ER-ANO-2002-1078-015 were evaluated by ANO Calculation 021381E101-212, "ANO Unit 1 EOTSG Replacement Piping Materials Evaluation for LBB", Rev. 0, to verify that the B&WOG report remained bounding. The leak-before-break re-evaluation did not involve new or different methodologies than were described in the FSAR. No other evaluation methodologies that are described in the FSAR are revised or replaced by this modification.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-017

I. OVERVIEW / SIGNATURES

Facility: **ANO - Unit 1**

Document Reviewed: ER-ANO-2002-1078-018, ANO-1 SG/RVCH Replacement - Interference Removal and Replacement **Change/Rev.: ERCN-1**

System Designator(s)/Description:

Description of Proposed Change

ER-ANO-2002-1078-018, ANO-1 SG/RVCH Replacement Project – Interference Removal and Replacement, provides for the removal of miscellaneous mechanical, electrical, and structural interferences that are not included in other Project ERs. Removal of these miscellaneous interferences is needed to facilitate the removal and replacement of the ANO-1 Once Through Steam Generators (OTSGs) and Reactor Vessel Closure Head (RVCH). Most of the removed interferences will be re-installed to their design configuration. Removal of two sections of Reactor Building (RB) facade and the relocation of the Reactor Building Purge system exhaust duct are the only two permanent plant changes. ER-ANO-2002-1078-018 also provides the basis for temporary alteration packages TA-05-1-005 and TA-05-1-006 per ANO Procedure 1000.028, "Control of Temporary Alterations".

Miscellaneous Mechanical Interferences

The interferences discussed below are inside and outside the Unit 1 Reactor Building. The interferences inside the Reactor Building are: 1) Reactor Coolant Pump (RCP) P32D Intermediate Cooling Water (ICW) System Supply pipe, 2) Breathing Air piping, 3) RCP Oil Collection piping, 4) Reactor Building Cooling System ductwork, 5) RCP P32B Drain Line Support and 6) Steam Generator E24A Hot Leg sample line. The interferences outside the Reactor Building are: 1) Penetration Room Ventilation System (PRVS) ancillaries, 2) Emergency Feedwater Initiation and Control (EFIC) System Steam Line 3/8" Instrumentation Tubing, 3) Reactor Building Purge (RBP) system exhaust duct located at azimuth 44°, 4) the RB façade at 30° and 270°, and 5) miscellaneous structural components.

Reactor Coolant Pump (RCP) P32D Intermediate Cooling Water (ICW) System Supply Pipe

Inside the Reactor Building in the area of P32D approximately 20' of pipe JBD-8-3" will be removed. This ICW piping provides cooling water to Backstop Lube Oil Cooler E-195D for the P32D lube oil system. This pipe section will be disconnected at a flange on one end and cut on the other. Each of the four openings of the pipe ends will be temporarily covered to prevent foreign material intrusion.

Breathing Air Piping

The Breathing Air system provides a reliable supply of dry, oil-free, high quality compressed air for use with inline
respirators throughout the plant. Portions of the Breathing Air system piping that will be temporarily removed are located inside the "D" ring of the E24A Steam Generator area. Approximately 5' of JDD-6-1" and one pipe support will be removed, stored and reinstalled. After removal of the pipe section, a pipe cap will be temporarily installed on the pipe stub at Floor Elevation 424' to allow for operation of the Breathing Air system as required during the SGR outage. The temporarily installed pipe cap must be capable of withstanding the system normal design pressure of 125 psig.

This system may be operated in its temporary configuration as required to support other operations and construction activities. Restoration will be accomplished when the design configuration of the pipe no longer interferes with the replacement of the steam generator. The Breathing Air system is non-safety related and there is no impact on any SSC by operating this system in a temporary configuration.

ER-ANO-2002-1078-018 and this 50.59 Evaluation provide justification for Temporary Alteration TA-05-1-005 which places the system in service in its temporary configuration and then restores the system to its design configuration.

Reactor Coolant Pump (RCP) Oil Collection

The upper and lower bearing housings are provided with an oil retention barrier. Oil that may leak during RCP operation is collected in the upper and lower retention barrier which drains to a collection tank. The two collection tanks (T-90 and T-91) are located in the Reactor Building on elevation 336'-6". Oil collected from P32A and P32B goes to T-91 and P32C and P32D goes to T-90.

Reinstallation of the RCP oil piping and associated supports will be to the ANO Unit 1 design configuration. No change is made to the pipe routing or size.

The Reactor Building Cooling System (RBCS) is safety related. The RBCS functions to provide cooling to the Reactor Building during power operation, plant shutdown and accident conditions.

Portions of the Reactor Building Cooling System return air ductwork and supports which interfere with the temporary Reactor Building construction opening (ER-ANO-2002-1078-007, Reactor Building Opening) will be removed. The duct and supports are non-safety related, however, they are Seismic Category II/I. The duct removal and replacement must be sequenced according to the requirements for the timing of the removal and replacement of the concrete and liner plate. The duct must be removed from the liner before the concrete removal process has reached a depth of 30" (i.e., remaining concrete depth ≥ 15"). The duct must not be supported by the liner plate during reinstallation until the cure of the reinstalled concrete has obtained a compressive strength of 3000 psi as required by ER-ANO-2002-1078-007, Section 4.2.2.

The RBCS will be operated during all modes including defueled to provide an environment in the Reactor Building which supports personnel in the performance of their activities. The RBCS must also be available to remove decay heat as required by the 1R19 SOPP.

The RBCS will be temporarily operated in an altered configuration to support the environmental conditioning inside the Reactor Building during activities associated with RVCH and OTSG replacement. DRNs are included for P&ID M-261 which show the system configuration with the duct removed. Calculation CALC-ANO-ER-04-033, "Evaluation of HVAC RBCS in the Interim Configuration for the Unit 1 SGRP", Revision 0 provides the allowance to operate the system in the temporary configuration with a portion of the return air duct removed.

In order to support Reactor Building cooling and the work required on the return air duct, at least one cooler must be operated with the inlet plenum door secured in the open position to allow air to flow across the chilled water coils but by-pass the return air duct. Other fans should be available as required by the 1R19 SOPP. Construction media roll type filters (or equivalent) will be temporarily installed either at the duct opening or the filter frame in the air handling units upstream of the chilled water coils to prevent particulate from entering the chilled water and service water coils for operation during the construction activities of the outage. These filters will be periodically checked and replaced as required to ensure cleanliness and prevent excessive pressure drop on the fans. Temporary Alteration TA-05-1-006 provides for the temporary filters and the altered configuration of the Reactor Building Cooling System.

The portion of the return air duct which will be temporarily removed is located between elevations 399'-6" and 426'-6" at azimuth 270°. The duct being removed is all welded construction with angle reinforcement. The cut areas of the duct sheet metal will be seal welded when reinstalled. The support angle will be cut from the baseplates which are attached to the Reactor Building liner plate. Upon reinstallation, the angle will be rewelded to the baseplates. No welding to the Reactor Building liner plate is required.

The Reactor Building Cooling ductwork and its associated supports will be reinstalled to ANO Unit 1 design configuration. All fans (VSF-1A, VSF-1B, VSF-1C & VSF-1D) must be shut down during duct sheet metal welding processes and must remain shutdown until the patches are completely installed where access holes were cut into the duct. The fans may only be restarted when the duct is completely closed with respect to air flow pressure boundary. Reinstallation may begin when the ductwork is no longer interference and must be complete prior to any SOPP requirements for refueling and entering Mode 4. However, transfer of the ductwork static weight load to the liner plate will not occur until construction opening repairs meet strength requirements defined in ER-ANO-2002-1078-007. Reinstallation of permanent supports and stiffeners may be accomplished with the fans operating provided that there exists no detrimental consequence to the welding procedure (vibration or air flow/negative pressure on sheet metal duct) or to personnel safety.

ER-ANO-2002-1078-018 and this 50.59 Evaluation provide justification for Temporary Alteration TA-05-1-006 which places the system in service in its temporary configuration and then restores the system to its design configuration.

Penetration Room Ventilation System (PRVS) 12" Exhaust Pipe Ancillaries

The Penetration Room Ventilation System (PRVS) filters air from the penetration areas in the event of penetration leakage from the Reactor Building during a loss of coolant accident. The PRVS is Seismic Category 1 and safety related. The PRVS initiates on an Engineered Safeguards Actuation System (ESAS) signal.

Technical Specification 3.7.11 requires the PRVS be operable in Modes 1, 2, 3, and 4. The PRVS is not required to be operable in Modes 5 and 6 and when the reactor is defueled. In order for the PRVS to be operable, among other requirements, Tech Spec Bases B3.7.11 c. requires that the ductwork be operable for the PRVS to be operable.

The PRVS will be affected by two areas of work: 1) the tubing for flow element FE-9836 will be temporarily removed and then reinstalled, and 2) the tubing for Isokinetic Probe AE-9835 will be temporarily removed and reinstalled. The insulation and heat tracing for the above sample tubing will be removed and reinstalled as required to support tubing removal and reinstallation.

All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled. All of the tubing, supports and insulation will be reinstalled per the design configuration prior to entering Mode 4. Replacement parts conforming to existing design specification or equivalent will be used as required. Compression fittings will be used for reinstallation of tubing.

Steam Generator E24A Hot Leg Sample Line

Approximately 17' of CCA-13-3/4" in the Reactor Building area around E24A will be temporarily removed along with 4 supports. This sample line is off of the Reactor Coolant Piping hot leg inlet to the steam generator. The pipe will be cut on one end and a fitting weld removed on the other. The supports will be removed by unbolting the intermediate connection plates. One snubber and two spring cans will be removed. Most hardware will be reused. Replacement bolt material conforming to or equivalent to design specification will be used if needed. The removed piping and the piping that remains will be closed on the ends to prevent foreign material/debris intrusion. FME and cleanliness controls will be per SGT Quality Execution Procedure 10.04, General Housekeeping, Cleanliness, and Foreign Material Exclusion (FME) Requirements. All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled.

Reinstallation will be per the design configuration except that an additional fitting may be used to connect the pipe where it was cut. The system will be restored prior to the plant entering Mode 4.

Steam Line 3/8" Instrumentation Tubing

The Emergency Feedwater Initiation and Control (EFIC) System (formerly - Steam Line Break Instrumentation and Control (SLBIC) System) is designed to protect against the consequences of a simultaneous blowdown of both steam generators. One section from each of the four 3/8" tubing lines will be temporarily removed in the area of the Reactor Building exterior buttress at azimuth 60°. All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 4.

RCP P32B Inlet Drain Line Support

The drain line support H002 is temporarily removed to eliminate an interference with SG E24B RCS cold leg temporary support. The RCS temporary support is installed and removed by ER-ANO-2002-1078-015. Support H002 will be disconnected from the RB floor by removing the nuts and washers from the anchors. Also the support will be disconnected from piping CCA-13-1½" by unbolting the pipe clamp from the sway strut rod end. The pipe clamp will remain attached to piping. All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 4.

Reactor Building Purge Exhaust Duct and Isokinetic Probe (AE-9820)

The Reactor Building Purge system serves to provide clean air in the Reactor Building during Modes 5, 6 and Defueled but cannot be operated during Modes 1 – 4 due to the inability of the Reactor Building isolation valves to close against DBA LOCA pressures. The exhaust air is sampled for radiological activity via an isokinetic probe located in the duct sections being relocated. Tubing connected to the probe is insulated and heat traced to prevent condensation of the air being transported through the tubing to the SPING system. Air is returned through tubing to the exhaust air duct upstream of the ER affected portion of the duct.

To accommodate SGRP tendon work, existing purge exhaust duct from fan VEF-15, located outside Unit 1 Reactor Building will be relocated from azimuth 44 to azimuth 30. This vertical section of ductwork has sample probe AE-9820 mounted in it at approximate elevation 482'-0". Probe AE-9820 is connected to ERGE-RMS (extended range gaseous effluent radiation monitoring system) monitor RX-9820 via ¾" tubing. This tubing is heat traced (refer to Electrical portion of ER-ANO-2002-1078-018 for details of heat trace removal). This heat tracing, along with the associated tubing and probe (AE-9820) will be relocated with the vertical ductwork. Insulation will be removed from sample probe AE-9820 back to and including the section of tubing to be removed. Heat tracing will be removed from sample probe AE-9820 back to and including the section of tubing to be removed. Heat trace cable will be rolled up and tied back out of the construction activity area. After the section of tubing that was removed has been re-installed, heat tracing and insulation will be re-installed.

Conduit EC2251 containing cables associated with the Reactor Building isolation valve CV-7401 will interfere with the relocated ductwork. This conduit needs to be removed, adjusted and reinstalled so that the conduit will not extend past 3" from the outside Reactor Building wall. This work is performed in the Electrical portion of ERANO-2002-1078-018.

Steel base plates will be attached to the RB wall in the location where the exhaust duct will be moved to during the outage. This activity will be performed prior to the outage with Unit 1 expected to be in Mode 1. This activity is addressed in a separate 10 CFR 50.65 Assessment.

Façade

The façade being removed is a painted sheet metal and steel structure which has no current system or plant function other than the aesthetics of the Reactor Building exterior. The façade is to be removed and disposed of as directed by ANO. This activity will be performed prior to the outage with Unit 1 expected to be in Mode 1 but may take place in any plant mode. This activity is addressed in a separate 10 CFR 50.65 Assessment.

<u>Miscellaneous Electrical Interferences</u>

RBCS Ductwork EL. 401'-6"

As previously discussed, the ductwork at 401'-6" elevation above the existing equipment hatch will be removed. The following conduits, temperature elements, and dew point elements are connected directly to the ductwork that is to be removed: Conduit J5379 (Cables I508B and I508D) feeds to HE-6278, Conduit J5380 (Cable I529D) feeds to TE-6278, Conduit J5381 (Cables I509B and I509D) feeds to HE-6279 and Conduit J5382 (Cable I530D) feeds to TE-6279. The cables will be de-terminated at the instrument and the associated cable will be pulled back and all conduits will be removed back to the first available junction box or cable tray that provides sufficient clearance for removal of the ductwork. The instruments will be removed and stored for reinstallation once the ductwork has been reinstalled. All of the above listed components are not safety related.

All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 4.

E24A Platform EL 424'-6" (Crow's nest)

To support the removal of the E24A platform steel and the RCS Hot Leg High Point Vent piping, four conduits that run along the structural steel and will be removed back to the fittings (LB condulets) at the north cavity wall. The conduits are oriented on the structural steel from top conduit to bottom conduit – SC2048, SC1079, SR2025 and J5241. Also conduits SC2046, SC2047, SC1077 and SC1078 and cables associated with the Valves SV-1082, SV-1084, SV-1081 and SV-1085 respectively are removed from the devices back to the LB condulet on the D-ring.

There is also a cable that is attached to building steel that is connected to TE-1080. There is an accessible plug near TE-1080. The cable will be unplugged and removed back to the D-ring and rolled up and tied back out of the rigging path.

All work on this system will be performed in Mode 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 5. The RCS Hot Leg High Point Vent piping will be removed and replaced by ER-ANO-2002-1078-015.

Jib Cranes (EL. 424'-6")

There are four disconnect switches with associated receptacles for jib cranes. The disconnect switches are S-101, S-102, S-103 and S-104. Only the conduits and associated cables for the disconnect switches (S-102, S-103, S-104) are required to be removed. The associated hoists will not be installed on these jib cranes. Switch S-101 is not affected by the ER.

All work on this system will be performed in any plant mode or while the reactor is defueled and will be restored prior to the plant entering Mode 4.

E24A Vibration Detection Element (VBE) EL. 392'-7"

Vibration and Loose Parts Monitoring System accelerometers are located on steam generator E24A. ANO-1 I&C Shop Technicians will perform the removal and reinstallation of the devices.

All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 4.

E24B Platform EL 424'-6" (Crow's nest)

To support the removal of the E24B platform steel and the RCS Hot Leg High Point Vent piping, four conduits that run along the structural steel and have to be removed back to the fittings (LB condulets) at the south north cavity wall. The conduits are, SC2042, SC1073, SR1021 and J5239. Also conduits SC2040, SC2041, SC1071 and SC1072 and cables associated with the Valves SV-1092, SV-1094, SV-1091 and SV-1093 respectively are removed from the devices back to LB condulet on the D-ring.

There is also a cable that is attached to building steel that is connected to TE-1090. There is an accessible plug near TE-1090. The cable will be unplugged and removed back to the D-ring and rolled up and tied back out of the rigging path.

All work on this system will be performed in Mode 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 5. The RCS Hot Leg High Point Vent piping will be removed and replaced by ER-ANO-2002-1078-015.

E24B Vibration Detection Element (VBE) EL. 392'-7"

Vibration and Loose Parts Monitoring accelerometers are located on steam generator E24B. ANO-1 I&C Shop Technicians will perform the removal and reinstallations of the devices.

Cable I332B will be de-terminated at VBE-1039B and pulled back through conduit J5215 to pull box TB 392. The cable will be rolled up and tied back out of construction activity. Conduit J5215 will be removed.

Cable I332F will be de-terminated at VBE-1039A and pulled back through conduit J5251 to pull box TB 392. The cable will be rolled up and tied back out of construction activity. Conduit J5251 will be removed.

All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 4.

Auxiliary Building Area 3 & 4 EL. 404’-0”

To support Reactor Building tendon activities, the following five conduits and associated cables in Auxiliary Building Areas 3 and 4 will be removed:

B5114 and B5116 - RB Tendon Gallery Recirculation Fan

EC2251 – RB Purge Outlet valve

EC2813 – EFW Pump Turbine steam admission

EJ2026 – Steam generator E24B Pressure Control

All work on these systems will be performed in Modes 5 and 6 or while the reactor is defueled. The systems will be restored prior to the plant entering Mode 4.

Reactor Building Tendon Gallery Supply Fan VSFM 23C, VSFM 23D

Cable B4151A (for fan VSFM 23C) and cable B4152A (for fan VSFM 23D) will be de-terminated at the fan motor. Fans VSFM 23C and VSFN 23D will be removed and stored for later reinstallation. There are no mode restrictions for these activities.

Miscellaneous Electrical Interferences E24A and E24B

To ensure protection of plant equipment, ANO-1 I&C Shop Technicians will perform the removal and reinstallation of SG E24A RTDs TE-2661, TE-2662, TE-2664/2665 (dual element RTD), SG E24B RTDs TE-2611, TE-2612, TE-2614/2615 (dual element RTD) and SG E24A shell thermocouples TE-2654, TE-2655,TE-2656, TE-2657 and TE-2658, SG E24B shell thermocouplesTE-2604, TE-2605, TE-2606, TE-2607 and TE-2608. The work will include the removal/determination and pull back of wiring prior to the removal of the Original OTSGs, as well as re-installation and re-termination of these devices after the replacement of OTSGs work has been completed. ANO-1 I&C will provide for safe storage of these components until the outage has progressed to the point where they can be reinstalled. ANO-1 I&C Shop Technicians will perform the work.

Ground straps embedded in the concrete base pedestal are connected to the skirt of the OTSGs (E24A and E24B). The ground straps are bolted to the skirt. The ground straps will be disconnected from the Original OTSG’s skirts and will be reconnected to the replacement OTSG’s skirts.

On the north side of the OTSG E24A TOS EL 392’-11½”, a # 4/0 AWG ground conductor is clamped to the web of the east-west W14X78 beam. The beam is going to be removed as a structural interference. The ground conductor will be removed from the beam clamps and split screw splice connections and secured to building steel on the north wall. This ground will be replaced after the steel is replaced.

Miscellaneous Electrical Interferences Outside the Reactor Building

The Super Particulate Iodine Noble Gas (SPING) monitor tubing for AE-9835 and the associated heat tracing will be temporarily removed and reinstalled at approximate elevation 437’-0” as part of the mechanical portion of this ER. The tubing will interfere with the tendon removal/replacement equipment. Probe AE-9835 is connected to monitor RX-9835 via this ¾” tubing. Electrically, the heat tracing system for probe AE-9835 consists of one circuit (circuit no.13) and two (2) thermocouples.

The Heat Tracing associated with the EFIC System tubing MS-54, MS-56, MS-58 and MS-60 is required to be removed temporarily to facilitate mechanical work and to be reinstalled after the completion of the mechanical work.

All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 4.

Miscellaneous Structural Interferences

Steam Generator Cavity Platform Removal

A motion study was used as the basis for determining what structural components are considered interferences within the steam generator cavities. These interferences were confirmed in a pre-replacement outage walkdown. Based on the motion study and walkdown, the cavity platforms at approximate elevations 393 ft and 384 ft will be partially disassembled to provide the required clearances. The remaining platforms at approximate elevations 347 ft and 376 ft clear the spatial envelope and do not require modification. Grating on the affected portions of the platforms will be tagged and removed until the support steel is reinstalled. The portions of grating remaining will be qualified for 100 psf live load. No shielding loads are permitted on the structural steel in steam generator cavities at approximate elevations 376', 384', and 393' during the duration of the modified configuration without engineering approval. The temporary platform configurations are qualified to ANO Seismic Category II/I criteria, and are therefore acceptable for plant operating Modes 5 and 6 and while the reactor is defueled.

Steam Generator Cavity Platform Reinstallation

Reinstallation of the platforms is in accordance with plant requirements including the AISC Manual of Steel Construction 6th Edition and the AWS Structural Welding Code, AWS D1.1, 1992. All structural bolted joints that are disassembled use new bolting material for reinstallation of the removed steel. Structural steel members that are cut for removal are welded back using full penetration welds. Therefore, the platforms are reassembled to their design condition. There are no permanent changes to the platforms other than the structural steel welding. All structures will be restored prior to entering Mode 4.

Elevation 424'-6" Steam Generator Cavity Decking Steel and Jib Cranes

The associated floor grating and two of the 33WF floor beams in each steam generator cavity will be removed. Three of the four RCP jib cranes (L32B, L32C, & L32D) are attached to the 33WFs and will be removed to provide clearance for the steam generator replacements. The jib cranes will be partially disassembled by removing the jib arms. The RCS Hot Leg High Point Vent system is mounted below the two 33WF beams along with the platform at elevation 419'-7½". The jib cranes, the 33WFs, the platform at elevation 419'-7½", and the RCS Hot Leg High Point Vent piping will all be removed as one unit. In order to accomplish this, W4 beams will be attached to the 33WFs to assist in the rigidity of the whole unit. After removal, the steel is stored in a reserved location on the 424'-6" elevation. One W10 beam that spans from the 419'-7½" platform framing to the cavity wall will be cut near the wall to provide a means for its removal. This W10 beam will be temporarily supported by the temporary W4 beam. The qualification of the entire unit is included with the structural interferences calculation (CALC ANO-ER-04-031). The 33WF beams are removed by unbolting. Reinstallation of these beams requires bolting the 33WFs using new bolts, welding the W10 with full penetration welds, removal of the W4s, and reassembling the jib cranes. Upon reassembly, the structures will be restored to their design condition which will occur prior to entering Mode 4.

Removal and reinstallation of the 33WFs will be coordinated with ER-ANO-2002-1078-015 which provides instructions for the removal and replacement of the RCS Hot Leg High Point Vent system. Also, temporary handrails must be provided as required for personnel safety at locations where the floor grating has been removed.

Elevation 401'-6" 12WF Floor Beam

The 12WF floor beam located parallel to the west side of the reactor building equipment hatch will be removed along with its associated grating to provide additional access for removal of the Reactor Building Cooling System air return duct. Removal of the beam provides approximately 2 feet of additional horizontal movement for the bottom of this duct after the duct is cut. The duct will be cut and removed as part of the mechanical portion of the ER. The 12WF beam will be removed and reinstalled by bolted connections. New bolting is used for the reinstallation. The beam will be restored to its design condition and the grating replaced prior to entering Mode 4. No qualification is required for the removal of this beam since its primary function is to resist floor loadings and its contribution to the overall stability and strength of the reactor building structural steel is insignificant.

Elevation 430'-6" Purge Valve Access Platform

The elevation 430'-6" access platform for the Reactor Building purge exhaust duct and valve CV-7403 will be partially disassembled to provide additional laydown area for the steam generator cavity decking floor beams. Approximately 3 feet will be removed from the outside face of this platform to provide the required space. The platform will be removed by unbolting and cutting two W8 beams. Reinstallation of the platform requires new bolting for the reassembled joints and welding the W8 beams with full penetration welds. Thus, upon reassembly, the platform is restored to its design condition. Barriers will be provided in the field to limit access to this platform during the time it is disassembled. No qualification is required for the partial removal of this platform since the vertical bracing and structural support points remain intact. The structure will be restored prior to entering Mode 4.

ORVCH Steel Removal

The structural interferences are removed from the ORVCH in the reactor building as required for rigging and handling, transport, and storage spatial requirements. The items removed for disposal include:

- The control rod drive mechanism (CRDM) cooling duct. This includes all the duct that normally remains attached to the service structure for movement to the head stand.
- The CRDM cooling duct supports. These supports are attached to the service structure steel, and are cut back to allow for the spatial requirements.
- The reactor head high point vent pipe supports, piping, and valves. These items are mounted on the service structure platform, and are cut back close to the top of the platform decking. Note that the PT-1070 instrument and its support, which are part of this system, are salvaged for reuse by ER-ANO-2002-1078-0639-000.
- The service structure platform handrail. The handrail is removed just above the toe plate.
- The ORVCH lifting pendants. The lifting pendants and associated hardware shall be carefully removed and stored as a contingency for any problems experienced with the new lift pendants and associated parts.

The ORVCH steel removal will be coordinated with ER-ANO-2002-1078-017 which addresses the radiation control measures needed to prepare the ORVCH for movement to the OSGSF. The remaining steel on the ORVCH cannot have sharp edges or projections that could cause damage to coverings that are to be used for contamination control.

The ORVCH steel removal must also be coordinated with ER-ANO-2002-1078-011 which addresses the design, temporary installation, and use of construction aid equipment to perform rigging and handling activities inside the reactor building. SGT will install a special lifting device for the ORVCH which will be used in the downending process required to move the ORVCH through the reactor building wall construction opening. In addition to ensuring that clearances are provided for the lifting device itself, the spreader beam used as part of the ORVCH lift rigging must pass over the service structure platform during the downending process, and requires clearance above the platform decking.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-017 Rev. 2)**	**Sections I, II, and IV required**

Preparer: Wayne R. Wasser / **ORIGINAL SIGNED BY WAYNE WASSER** / Adecco Tech / SG/RVCH / 9-11-05
Name (print) / Signature / Company / Department / Date

Reviewer: Doyle G. Adams / **ORIGINAL SIGNED BY DOYLE ADAMS** / EOI / SG/RVCH / 09-12-05
Name (print) / Signature / Company / Department / Date

OSRC: J.R. Eichenberger / **ORIGINAL SIGNED BY J.R. EICHENBERGER** / 9-15-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	ANO-1 FSAR Figure 9-14 (P&ID M-218 sheet 5) as part of TA-05-1-005 and ANO-1 FSAR Figure 6-7 (P&ID M-261, sheet 1) as part of TA-05-1-006 will be temporarily affected but not permanently changed
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR? ☐ **Yes** ☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. Basis

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

In support of replacement of the Unit 1 steam generators and RVCH, a number of mechanical, electrical, and structural interferences will be moved to eliminate the interference. Each interference will be removed in a Mode when it is not required to be in service. Each interference will be returned to its design configuration with the exception of two sections of the RB facade which will not be replaced.

Operating License/Technical Specifications (TSs)

Of the mechanical/electrical/ structural interferences that will be removed and later replaced, only the Reactor Building Purge System, Reactor Building Cooling System, Penetration Room Ventilation, and EFIC Systems are discussed in the OL/Tech Specs. The Intermediate Cooling Water, Breathing Air, SG Hot Leg Sample, cold leg drain, Jib Cranes, Vibration and Loose Parts Monitoring System, ground strap, CRDM Cooling, shielding monorail, Reactor Building façade, Reactor Building tendon gallery fans, Super Particulate Iodine Noble Gas monitor, RCP Oil Collection System, and Reactor Building Platforms are not discussed in the Operating License or Technical Specifications. They are not discussed in the Tech Spec Bases, Technical Requirements Manual, or Core Operating Limits Report.

The Reactor Building Purge System is discussed in Tech Spec Sections 3.6.3 and 3.9.3; TS Bases Section 3.6.3 and 3.9.3; and TRM Section 3.6.1. No changes to the Tech Specs, Tech Spec Bases, or TRM are needed.

The Reactor Building Cooling System is discussed in Tech Spec Sections 3.3.6 and 3.6.5; TS Bases Section 3.3.5, 3.3.6, 3.3.7, 3.6.5, and 3.7.7; and TRM Section 3.6.5. No changes to the Tech Specs, Tech Spec Bases, or TRM are needed.

The Penetration Room Ventilation System is discussed in Tech Spec Sections 3.7.11 and 5.5.11 and in Tech Spec Bases Sections 3.3.5, 3.3.6, 3.3.7, and 3.7.11. No changes to the Tech Specs or Tech Spec Bases are needed.

The Emergency Feedwater Initiation and Control (EFIC) System is discussed in Tech Spec Sections 3.3.11, 3.3.12, 3.3.13, 3.3.14 and, 3.8.9. It is discussed in Tech Spec Bases Section 3.3.5, 3.3.6, 3.3.7, 3.3.11, 3.3.12, 3.3.13, 3.3.14, 3.7.2, 3.7.3, 3.7.5, 3.8.7, and 3.8.9. No changes to the Tech Specs or Tech Spec Bases are needed.

TRM 3.4.3 limits steam generator secondary side pressure to <200 psig with the steam generator shell temperature <100°F. The steam generator shell thermocouples are used to show compliance with this requirement. With the secondary side of the steam generator piping cut for steam generator replacement, pressurization of the secondary side of the steam generators to >200 psig is not possible. The thermocouples will be returned to an operable status prior to pressurization of the ROTSGs. No changes to the TRM are needed. The SG Downcomer and Downcomer Inlet RTDs are not discussed in the Tech Specs, Tech Spec Bases, or TRM.

The RCS high point vent is discussed in TRM Section 3.4.2. No changes are required.

FSAR

The following systems that are impacted by this ER are discussed in the FSAR. The function of each system remains unchanged therefore no changes to the FSAR are needed.

The Intermediate Cooling Water (ICW) System is discussed in various locations in the FSAR. No changes to these sections are needed. This portion of the ICW System is not in service during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

The Breathing Air (BA) System is discussed in various locations in the FSAR. No permanent change to the FSAR is needed. FSAR Figure 9-14 shows the BA System. During the SGR outage, a section of the BA System piping will be removed and the remaining piping will be capped as part of TA-05-1-005 so that the BA System can be returned to service. TA-05-1-005 includes temporary changes to FSAR Figure 9-14. Prior to completion of the SGR outage, the BA System will be returned to its design condition. Changes to this system are temporary and the system will be restored to design condition.

The RCP Oil Collection System is discussed in various FSAR Sections. No changes to these sections are needed. The RCP Oil Collection System is not in service during the SGR Outage. Changes to this system are temporary and the system will be restored to design condition.

The Reactor Building platforms are discussed in FSAR Section 5.1.7.1. No change to this section is needed. Changes to the platforms are temporary and will be restored to their design condition.

The Reactor Building Cooling System (RBCS) is discussed in a number of locations in the FSAR. These sections describe the operation of the RBCS during normal (power) operation and in post-accident conditions. The planned temporary changes (TA-05-1-006) to the RBCS will not affect the operational modes that are described in the FSAR. TA-05-1-006 includes temporary changes to FSAR Figure 5-7. Prior to completion of the SGR outage, the RBCS will be returned to its design condition. Changes to this system are temporary and the system will be restored to design condition.

The Reactor Building Purge System is discussed numerous locations in the FSAR. No changes to these sections are needed.

The RCS hot leg sample is discussed in FSAR Section 5.2.2.4.1. No changes to this section are needed. The RCS hot leg sample is not in service during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

The Penetration Room Ventilation System (PRVS) is discussed in various locations in the FSAR. No changes to these parts of the FSAR are required. The PRVS is not required to be operable during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

The Emergency Feedwater Initiation and Control (EFIC) System is discussed in various FSAR Sections. No changes to these sections are needed. EFIC is not required to be operable during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

The Jib Crane is discussed in FSAR Section 9.6.1.7.1. No changes to the FSAR are needed. Changes to the Jib Cranes are temporary and will be restored to their design condition.

The RCS high point vent is discussed in various parts of the FSAR. No changes to the FSAR are needed. The high point vent is not required to be operable during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

The Vibration and Loose Parts Monitoring System are discussed in various parts of the FSAR. No changes to the FSAR are needed. The Vibration and Loose Parts Monitoring System is not required to be operable during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

The Super Particulate Iodine Noble Gas (SPING) Monitor System is discussed in various parts of the FSAR. The SPING Monitor System is not required to be operable during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

Steam Generator Shell Temperature indication is discussed in various parts of the FSAR. No changes to these sections are needed. The Steam Generator Shell Temperature indication is not required to be operable during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

Test or Experiment Consideration

Removal and reinstallation of mechanical, electrical, and structural interferences in support of steam generator and RVCH replacement will remove a number of systems from service. Each of these systems with the exception of two sections of the RB façade, will be returned to its design or modified configuration. Each system that is removed from service will be appropriately tested prior to return to service. None of the systems returned to service will be used for functions that are outside of their design basis. Removal and reinstallation of these interferences is not considered a test or experiment.

Platforms, handrails, steel beams, and grating in specific locations, along with the jib cranes located at elevation 424 (nominal), have been identified as interferences inside the Reactor Building. These components will be removed and later returned to their design configuration. Removal of these structures is not considered a test or experiment.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search: LRS 50.59 – Unit 1	Keywords: “intermediate cooling water”, ICW, “breathing air”, “oil collection”, “Containment Cooling”, “Reactor Building Purge”, “Reactor Building Cooling”, “RB Cooling”, “hot leg” w/20 sample, “reactor coolant system” w/20 sample, platform*, grating*, “penetration room ventilation”, PRVS, “Emergency Feedwater Initiation”, EFIC*, jib, “loose parts”, ground w/20 strap*, “steam generator” w/20 ground, CRD* w/5 cooling, “control rod” w/5 cooling, “high point”, monorail, “hydrogen purge”, “super particulate”, SPING*, “temperature sensor”, “temperature sensors”, “temperature element”, “temperature elements”, “temperature indicator”, “temperature indicators”, “pressure indicator”, “pressure indicators”, “flow element”, “flow elements”, L32*, L-32*, drain w/10 (pip* OR line*), “shell temperature”, “inlet temperature”, vibration w/20 detect*”, “tube to shell”, tube*to*shell, “SPDS”, “Safety Parameter Display”, ICS, “Integrated Control”, “vibration and loose parts”, “green train”, “red train”, HE-6*, E-33*, TE-6*, SV-108*, SV-109*, GCR*, RCR*, PT-10*, E-4*, E-252*, VBE*, TE-2*, RJI*, S-10*, TE-2*, E-7*, JB*, J53*, NJ*, SC1*, SC2*, GJI*, B532*, B44*, J52*, SC*1*, SC*2*, deck*, “building internals”, “nitrogen system”, “nitrogen supply”, thermocouple* w/20 (CRD OR control), (plate* OR clamp*) w/20 (CRD OR control), position indicator*, (duct* or “duct work”) w/20 (reactor OR containment), lub* near/10 (RCP OR “coolant pump”), SLBIC, “steam line break instrumentation”, M-*, ERG*, “extended range”, isokinetic, ae-*, (“tendon gallery” w/20 fan*), VSFM*, facade, architectural, fascia, pendant*, TE-10*, E-195*, E195*, T-90, T90, T-91, T-91, “lube oil”w/5 cooler*, ZS*, “SV-7401”, “I/P-2618”, VEF-15*, VEF15*, “CV- 7401”, EC2*, EJ2*, B511*, GC*, SV-10*, SR1*, crow*, VBE*
LBDs/Documents reviewed manually: ANO Unit 1 FSAR	Chapter 7 and Sections1.4.39, 4.2.2.5, 4.2.3.8, 4.2.4.7, 5.2.2.4.1, 5.2.4.4, 5.2.6, 5.2.7, 6.3, 6.5, 9.6.1.7.1, 6.6.2.1, 7.1.3.2.4, 9.6.1.1, 9.7, 11.1.3.4.1, 11.1.3.8, 13.6, 14.2.2.5.2.3, Tables 6-10 and 6-13, and Figures 5-7,6-4, 6-10, 6-13, 9-6, 9-7, 9-8, 9-10, 9-14, and 9-20.

5. **Is the validity of this Review dependent on any other change?** (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☐ **Yes**
☒ **No**

If “YES,” list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☒	☐	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☒	☐	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

Access to the Auxiliary Building roof area at Elevation 358'-0" (area between the Auxiliary and Reactor Buildings) will be required. Access will be via roof hatch number 54 and door number 482. Door 482 is an entrance portal from the outside to the inside of the plant via the Auxiliary Building. To prohibit entrance to this area of the plant, the door is bolted. Compensatory measures will be required during the time that this door is unbolted. Additionally this portion of the Auxiliary Building roof area should be considered a "confined space". Ongoing activities in this area should be coordinated and communicated with ANO Safety, RP, and Security to ensure compliance with applicable requirements.

D INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: _____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10 CFR 50.59 Review Program Guidelines for guidance.

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.	☐ Yes ☒ No

Does the proposed Change:

1.	Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?	☐ Yes ☒ No

BASIS:

The accidents that are discussed in the FSAR that are applicable in Modes 5 or 6 or are not mode related are: fuel loading errors (FSAR Section 14.1.2.10), fuel handling accident (FSAR Section 14.2.2.3), and waste gas tank rupture (FSAR Section 14.2.2.7). The fuel loading errors, fuel handling accidents, and waste gas tank rupture are unrelated to activities performed by ER-ANO-2002-1078-018. The activities in this ER (except removal of two sections of RB façade, installation of the RB purge base plates, and removal of the RB tendon gallery fans) occur during shutdown (Modes 5, 6, and defueled) and have no impact on the previously discussed FSAR accidents. Removal of the RB façade sections and tendon galley fans and installation of the RB Purge base plate will be performed outside of the Reactor Building and will have no impact on previously evaluated accidents. The RB façade and RB Purge base plate activities are addressed in a separate 10 CFR 50.65 Assessment.

Therefore, the implementation activities of this ER will not increase the frequency of any accident previously evaluated in the FSAR where power or hot operation (Modes 1-4) is assumed.

2.	Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?	☐ Yes ☒ No

BASIS:

A number of the interferences that will be temporarily removed by this ER perform safety functions during normal operating modes. Work on these systems will not begin until the Unit is in a mode where the system is not required by the Tech Specs or TRM, as applicable, and the system has been released for work by EOI. Upon restoration, each SSC will be restored to its design configuration.

Therefore, none of the planned activities will increase the occurrence of a malfunction of an SSC important to safety previously evaluated in the FSAR.

3.	Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR?	☐ Yes ☒ No

BASIS:

ER-ANO-2002-1078-018 temporarily removes and replaces a number of mechanical, electrical, and structural interferences that function to mitigate the consequences of accidents previously evaluated in the FSAR. None of the SSCs that are part of this ER are accident initiators. The function of some of the components is to mitigate or assess accidents. These SSC will only be removed when they are not longer required to perform their safety related function. These SSCs will be restored to meet or exceed its original design condition.

Therefore, the removal and replacement of the interferences described in ER-ANO-2002-1078-018 does not result in any increase in the consequences of an accident previously evaluated in the FSAR. It can be further stated that there is no more than minimal increase in the dose expected from an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

With the exception of the RB façade removal,SSCs that are temporarily removed as part of ER-ANO-2002-1078-018 will be restored to their design or modified condition. This ER does not include any changes to the operating characteristics or functions of any affected SSCs. Likewise, there are no changes to the design basis for any of the affected SSCs. In essence, each SSC will be returned to an equivalent condition both physically and operationally.

Additionally, during Modes 5 and 6, partially disassembles SSCs will be left in a condition so as not to create a Seismic II/I condition to SSCs that are required for these modes.

Therefore, the removal and replacement of the interferences described in ER-ANO-2002-1078-018 does not result in any increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR. It can be further stated that there is no more than a minimal increase in dose expected from a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

ER-ANO-2002-1078-018 removes a number of mechanical, electrical, and structural interferences. Interference removal activities will not begin on each system until it is no longer required to be operable and is released by ANO-1 Operations. Each system will be returned to its design configuration with the exception of the RB façade. Two sections of the façade on the exterior of the Reactor Building will be removed and not replaced. The Reactor Building Purge exhaust duct will be relocated from azimuth 44° to azimuth 30° and will be installed to its original design conditions.

Therefore, the planned interference removals will not create the possibility for an accident of a different type than previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

ER-ANO-2002-1078-018 removes a number of mechanical, electrical, and structural interferences in support of the SG/RVCH replacement. All SSCs (except two sections of the RB facade) will be restored to their design or modified configuration. After restoration, all SSCs will remain qualified to perform their function in normal conditions and all applicable accident conditions. ER-ANO-2002-1078-018 does not change any design, system or functional parameters of each affected SSC.

Therefore, ER-ANO-2002-1078-018 will not create the possibility for a malfunction with a different result than previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

ANO-1 fission product barriers include fuel cladding, RCS Boundary, and the Reactor Building Pressure Boundary.

<u>Fuel Cladding</u>

The planned removal of interferences inside the Reactor Building will not begin until the Unit is in Modes 5, 6, or the core has been defueled. The only activity that could be a concern for loose parts entering the reactor is the RCS hot leg sample line removal. FME practices and procedures will be followed to prevent the introduction of foreign material into the RCS.

RCS Boundary

The removal of the RCS hot leg sample line is the only interference that impacts the RCS pressure boundary. The line will be removed with the Unit in Mode 5 or 6 or with the all fuel removed from the reactor vessel and when released by ANO-1 Operations. It will be returned to its design configuration prior to filling the RCS and entry into Mode 6. All work on the system will be in accordance with ANSI B31.7. The integrity of the RCS pressure boundary will be verified by conducting an inservice leak test in accordance with ASME Section XI.

Reactor Building Pressure Boundary

The removal and replacement of SSCs describe in ER-ANO-2002-1078-018 will not adversely impact the pressure retaining capability of the Reactor Building or other fission product barriers. Supports for the Reactor Building Cooling System return ductwork will be welded to the existing base plates. No welds will be made to the Reactor Building Liner Plate.

Therefore, the activities describe in ER-ANO-2002-1078-018 do not affect the design basis limit of the fission product barriers as described in the FSAR being exceeded or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

No new methods of evaluation were used in the temporary relocation of mechanical, electrical, and structural interferences in support of the Unit 1 SG/RVCH replacement outage or in the permanent removal of two sections of the façade and relocation of the Reactor Building Purge exhaust duct on the exterior of the Reactor Building.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-018

I. OVERVIEW / SIGNATURES

Facility: ANO - Unit 1

Document Reviewed: ER-ANO-2002-0639-000, ANO-1 Service Structure Replacement **Change/Rev.:** 0 ERCN-13

System Designator(s)/Description: RCS, RVCH, CRDM

Description of Proposed Change:

ER-ANO-2002-0639-000, "ANO-1 Service Structure Replacement", proposes to replace the original Control Rod Drive Service Structure (CRDSS) with a new CRDSS that will be attached to the replacement CRDSS support skirt installed by ER-ANO-2002-0638-000, "ANO-1 Reactor Vessel Closure Head (RVCH) Replacement". The replacement CRDSS is a modification that will enhance the capabilities of the CRDSS. The enhancements are to reduce disassembly and reassembly effort and time, thus reducing personnel exposure and simplifying component handling to improve personnel safety. Also modifications will be made to the lifting rig assembly to accommodate the replacement CRDSS design and a Tripod Storage Stand will be installed inside containment.

Changes addressed by this review:

- The replacement CRDSS is comprised of upper and lower service structure cylinders in place of the existing single cylinder. With the two cylinders bolted together and the CRDM platform welded to the upper cylinder this fully replaces the main components of the existing CRDSS.
- The design of the replacement CRDSS Batwings allows the Batwings to remain attached to the CRDSS platform while the RVCH and CRDSS assembly is moved. Two actuators per Batwing are operated simultaneously to rotate the Batwings after all the appropriated cables have been disconnected.
- The existing CRDSS was classified as non-safety related and the ASME B&PV Code was not imposed for the design or construction. The replacement CRDSS is safety related and designed to ASME B&PV Code Section III except for the hardware above the work platform deck plates, service air system, shielding and stud tensioner monorails and Batwings which are non-safety related components and/or structures.
- The three existing 2-ton air operated chain hoist and trolleys will be replaced with three new 6600-lb (3 metric tons) lifting capacity air operated chain hoist and chain driven trolleys. The loading chain will be stainless steel instead of galvanized which will cause the lifting capacity to be de-rated to 4400-lb (2 metric tons). The hoist and trolleys will remain attached to the monorail during plant operation.
- The Intermediate Cooling Water (ICW) headers have been redesigned to facilitate ease of connection and disconnection of the tubing spool pieces (flex hoses) used for supplying the cooling water to the CRDMs. The redesigned headers will have fewer penetrations with each penetration servicing more CRDMs.
- There will be eight (8) 10 inch-by-12 inch ventilation holes in the lower section of the CRDSS to accept the connection of the redesigned ductwork. The existing CRDSS utilized eleven (11) 12-inch diameter and one (1) 12 inch-by-18 inch elliptical ventilation ports.
- The CRDSS will provide support for the newly designed ventilation plenum, which is designed as an integral part of the CRDSS.
- An elliptical opening measuring 14 X 18-inches whose centerline is located approximately 20 3/8 inches from the bottom mounting flange of the CRDSS.
- To enhance the ability for future inspections, inspection ports are located vertically on center at 8.88 inches from the lower cylinder's bottom flange. There are 12, 2-inch by 5-inch slotted ports and 16, 2-inch diameter circular ports, which are located to provide enhanced examination and inspection capability of the RVCH and CRDMS.
- A full circumference band is provided to cover the elliptical openings and the smaller inspection ports when they are not being used.
- The existing support structure for the temporary radiation shielding blankets is attached by a banding configuration with the foot of the structure on the CRDSS lower flange. The support structure on the replacement CRDSS will be welded to the lower CRDSS cylinder assembly above the ventilation ductwork ports.
- Service air manifolds are attached to two sides of the replacement CRDSS work platform.

- All carbon steel surfaces of the replacement CRDSS will be coated with Carboguard 890N with the exception of the areas identified in ER-ANO-2004-0631-000.
- The fiber braided ropes will be replaced with fixed pendants that will pass through similar openings in the platform floor plates and support structure.
- The ends of the fixed pendants will be threaded with the lower end threaded into a clevis for connection to the RVCH lifting lugs and the upper threaded into a turnbuckle. The other end of the turnbuckle will have a T-Lug threaded into it. All threaded connections will have a lock nut to ensure connections will not unthread during use.
- The T-Lug is designed to be used with the lifting fixture latch boxes.
- The existing braided ropes are only attached to the RVCH lifting lugs during plant shutdown activities that require RVCH movement. The replacement fixed pendant assemblies will remain attached to the RVCH lifting lugs during all modes of plant operation.
- When the fixed pendant assemblies are not attached to the lifting tripod, lateral movement is restrained by squeeze plates fastened to slotted cover plates to lock the pendants in place. The slotted cover plates are attached to the platform and provide a path for the pendants to move during lifting operations.
- The reactor vessel head vent and High Point Vent system will be replaced on the replacement RVCH/CRDSS with equivalent piping, supports and valves.
- There is a weight difference between the existing RVCH/CRDSS configuration and the replacement RVCH/CRDSS configuration when being lifted to head stand of approximately 68,000 lbs.
- The addition of a Tripod Storage Stand for permanent use inside the Containment Building.
- The internals handling adapters (IHAs) will be replaced with equivalent IHAs which are capable of a higher lift capacity.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☒	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-018)**	**Sections I, II, and IV required**

Input Provided by: Douglas Beckner / **ORIGINAL SIGNED BY DOUG BECKNER** / AREVA / NSSS / 3-31-05
Name (print) / Signature / Company / Department / Date

Preparer: Jerry Howell / **ORIGINAL SIGNED BY JERRY HOWELL** / EOI / SG-RVCH / 4-1-05
Name (print) / Signature / Company / Department / Date

Reviewer: Randall Smith / **ORIGINAL SIGNED BY RANDALL SMITH** / Universal Personnel / SG-RVCH / 4-1-05
Name (print) / Signature / Company / Department / Date

OSRC: J.R. Eichenberger / **ORIGINAL SIGNED BY J.R. EICHENBERGER** / 4-14-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	Table 4-12
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR? ☐ **Yes** ☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. Basis

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.3.2 for guidance.

This review addresses replacing the Control Rod Drive Service Structure (CRDSS) with a replacement-in-kind that will incorporate design enhancements and will be mounted on the replacement CRDSS support skirt. As part of this proposed activity, the reactor vessel closure head (RVCH) lifting rig assembly was re-designed to better interface with the new CRDSS. A Tripod Storage Stand will be mounted in the containment building to allow the RVCH lifting tripod, internal handling adapters and spreader ring to be stored together. The CRDSS support skirt and the mounting of the replacement CRDSS on the new support skirt is addressed in ER-ANO-2002-0638-000. Also the replacement CRDSS will accommodate the redesigned ventilation ductwork, which is addressed in ER-ANO-2003-0366-000, "ANO-1 CRDM and Service Structure HVAC Ductwork Modifications".

OPERATING LICENSE

The replacement CRDSS was designed and fabricated with the intent to maintain the current licensing basis and to accommodate plant life extension as necessary for continued operation (SPEC-ANO-M-579, "Equipment Design Specification for Replacement Control Rod Drive Service Structure ANO-1"). Although the replacement CRDSS provides some enhancements that will provide future benefits for the plant, the functions as described in the FSAR will still be accomplished.

FSAR

As stated in FSAR Section 4.3.3, "The control rod drive service structure is designed to support the control rod drives to assure no loss of function in the event of a combined LOCA and Maximum Hypothetical Earthquake (MHE)." The replacement CRDSS will adequately provide this function.

The following changes will be required to the ANO-1 FSAR:

- Section 9.6.1.6, Stud Tensioner & Tooling Hoist, change to address the description of the replacement stud tensioner hoist.
- Section 9.6.1.6, Head Lifting Device, the fiber braided rope will be replaced with a description of the replacement fixed pendant assembly (FPA).
- Section 9.6.1.6, Internals Handling Adapter, change to identify the use of adapter for RVCH lift.
- Section 9.6.2.2, change from referencing two drawings for lifting rig description to referencing one drawing.
- Figure 9-48 will change to address replacement head lifting device.
- Figure 9-49 will change to address the replacement closure head removal lift configuration.
- Figure 9-49A will be deleted to provide only one figure to represent the closure head lift configuration.
- A figure depicting the Tripod Storage Stand will be added to the FSAR.

In addition, FSAR Table 4-3 was revised by ER-ANO-2002-0638-000, "ANO-1 Reactor Vessel Closure Head (RVCH) Replacement" to reflect the weight of the replacement RVCH. This additional weight was combined with the additional weight of the replacement CRDSS and the total weight addition was evaluated in CALC-020639E101-01 an addressed in this review.

The following activities will impact the CRDSS or lifting assembly design and are reviewed in Section III – 50.59 Evaluation Exemption:

- There will be an upper and a lower service structure cylinder (ANO-1 Dwgs.M1B-507 and M1B-510) that will be aligned and bolted together.
- New designed Batwings (ANO-1 Dwgs.M1B-553, M1B-554, M1B-555, M1B-556, M1B-557, and M1B-558) to remain attached to the CRDSS platform while the RVCH and CRDSS assembly is moved.
- The new CRDSS (ANO-1 Dwg. M1B-500) was designed and fabricated safety related.
- Three new 6600-lb (de-rated to 4400-lb with stainless steel load bearing chains) lifting capacity air operated chain hoist and chain driven trolleys to remain attached to the stud tensioner hoist monorail during plant operation.
- New designed Intermediate Cooling Water (ICW) supply and return headers (ANO-1 Dwgs. M1B-567 and M1B-568).
- Different designed ventilation holes (ANO-1 Dwg. M1B-510) to connect newly designed cooling air ductwork.
- New personnel and equipment access opening (ANO-1 Dwg. M1B-510) to the interior CRDSS chamber and the cover (ANO-1 Dwg. M1B-510) provided for plant operation.
- There are 12, 2-inch by 5-inch slotted ports and 16, 2-inch diameter circular ports for examination and inspection of the RVCH and CRDMS (ANO-1 Dwg. M1B-510).
- A full circumference band (ANO-1 Dwg. M1B-532) is provided to cover the inspection ports during plant operation.
- The support structure for the shielding blankets monorail (ANO-1 Dwg. M1B-521) will be welded to the lower CRDSS.
- Service air manifolds are attached to the replacement CRDSS work platform (ANO-1 Dwg. M1B-570).
- All carbon steel sections of the replacement CRDSS will be coated with Carboguard 890N with exception of those areas identified in ER-ANO-2004-0631-000.
- Squeeze plates (ANO-1 Dwg. M1B-614) are provided to provide lateral support for the fixed pendants when rods are not attached to tripod.
- The existing RV vent line and High Point Vent (HPV) system will have the electrical connections disconnected and the mechanical portions will be discarded or salvaged. New piping materials, supports and solenoid valves will be erected to the replacement RVCH/CRDSS assembly (ANO-1 Dwg. M1B-500) prior to moving the assembly into containment.
- The three braided wire ropes will be replaced with fixed pendant assemblies (Fixed Pendant Assembly, FPA, ANO-1 Dwg. M1B-612).
- The fixed pendant will be threaded with the lower end threaded into a clevis for connection to the RVCH lifting lugs and the upper threaded into a turnbuckle (ANO-1 Dwg. M1B-612).
- The other end of the turnbuckle will have a T-Lug threaded into it (ANO-1 Dwgs. M1B-612 and M1B-616).
- All threaded connections will have a lock nut to ensure connections will not unthread during use (ANO-1 Dwg. M1B-612).
- The T-Lug is designed to be used with the lifting fixture latch boxes (ANO-1 Dwgs. M1B-612 and M1B-615).
- Once assembled the three Fixed Pendant Assemblies will remain attached to the RVCH lifting lugs during all modes of plant operation.
- The internals handling adapters (including lift fixture latch boxes) will be replaced with like internals handling adapters which are rated for increased lift capacity (ANO-1 Dwg. M1B-621).

The following are reused and will not impact the CRDSS design and do not require further 10 CFR 50.59 review:

- The existing RADCAL assembly would be reused and installed on the replacement RVCH, nozzle location number one (1), core location H8. The location, mounting, operation and support function provided by the replacement RVCH and CRDSS is the same as the existing configuration therefore there will be no change to the RADCAL assembly or reactor vessel RADCAL parallel vent line.
- The existing seismic clamps and tie plates (ANO-1 Dwg. M1B-513) will be reused on the replacement CRDSS in the same configuration as the existing CRDSS therefore there would be no change to the lateral restraint function with the replacement CRDSS.

- The existing power and instrument cables and connections will be reused on the replacement CRDSS. The new Batwing design provides the same routing and support for the cables as the existing CRDSS.
- The Loose Parts Monitoring (ANO-1 Dwgs. M1B-526 and M1B-527) will be re-attached to the replacement CRDSS as similar to the existing configuration as possible.
- The nine (9) thermocouple terminal boxes (ANO-1 Dwg. M1B-500) will remain attached to the existing mounting platform used by these boxes, which will be removed and reattached to the replacement CRDSS platform assembly. This terminal box platform will be located and oriented in the same manner as currently, thus allowing reuse of instrument cables and maintaining the same capabilities and functionality.
- The tripod will be relocated and stored on the Tripod Storage Stand during plant operation.

The following changes could reduce the level of conservatism used in various SSC evaluations and therefore are being reviewed in Section IV – 50.59 Evaluation:

- The additional combined weight of the RVCH and CRDSS modifications impact on the Reactor Coolant System (RCS) supports and the reactor internals (ANO-1 CALC-020638E101-17).
- The increase in surface area and volume introduced by the replacement RVCH and CRDSS on the margins for the post accident containment peak pressure analyses (ANO-1 CALC-020638E101-20).
- The installation of the Tripod Storage Stand will increase the surface area and reduce the containment free volume and thus impact the margins use in calculating the post accident containment peak pressures (ANO-1 CALC-020638E101-20).

No operational procedure described in the FSAR is affected by this activity. The change involved in this proposed activity is replacing the CRDSS with a new fabricated CRDSS. The replacement CRDSS would perform the same functions as the existing CRDSS and also is designed so that all connecting or supporting components would continue the same functionality without change. The Tripod Storage Stand (TSS, ANO-1 Dwg. M1B-601) will allow secure storage during operation of equipment used only during refueling outages. Since the TSS will remain inside of the containment during operation, the TSS was evaluated (ANO-1 CALC-020639E101-11) for Seismic II/I and operating conditions. Based on the results of the calculation it is concluded that the TSS will not compromise the functional capability of any other SSC within the containment.

The proposed activity does not involve revising evaluation methodology, but does involve providing a component designed, fabricated and tested to ASME Code requirements and owner's approved requirements. This would ensure that any input or assumption used in any evaluation methodology to define the design bases and safety analyses would not be impacted by this activity.

The intent of all operational procedures is maintained and the activity does not involve a change to any procedure that would adversely affect how any SSC performs or controls any design function as described in the FSAR.

No operational limits are changed, no parameters altered and no additional operational requirements are evoked by this proposed activity.

<u>OTHER LBDs</u>

All the ANO-1 Technical Specifications will remain applicable with no changes. The replacement CRDSS was designed to meet the current Technical Specifications (TS). There will be neither reduction in margins or changes to any currently applicable limits nor any new restrictions or limits imposed by the replacement CRDSS.

There will be no change to the TS Bases, Technical Requirements Manual, Core Operating Limits Report, Offsite Dose Calculations Manual or NRC Safety Evaluation Reports due to the proposed CRDSS replacement.

Question II.A.2 – This activity involves the addition of new components and the replacement and modification of existing plant components. All new, replacement and modified components have been thoroughly evaluated to determine their capability, applicability and performance. Therefore this activity does not involve any tests or experiments. All SSCs will be operated within the design limitations/requirements and within the limits described in the FSAR.

Conclusions: All the changes addressed in this review have been analyzed and evaluated to have positive, neutral or acceptable minimal adverse impacts. The difference in lift weight is within the capacity of the up rate on the reactor building polar crane. The functionality and capability of all structures, systems and components at ANO-1 are maintained or enhanced. All FSAR described design functions will continue to be available without any degraded ability. ANO-1 would continue to operate within the requirements of the current Operating License (OL) and Technical Specifications (TS), therefore this activity may be performed without prior Nuclear Regulatory review and approval.

4. <u>References</u>

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LRS 50.59 – Unit 1	“Reactor vessel”, “internal handling adapter”, “stud tensioner”, “tooling hoist”, “terminal box”, “Fiber Braided Rope”, containment, “reactor building”, “closure head”, CRDM, nozzle, “control rod”, ASME, “cooling water”, air, ventilation, “service structure”, batwing, coating, platform, pendants, “spreader ring”, tripod, HVAC, lifting, cables and accident

LBDs/Documents reviewed manually:
Arkansas Nuclear One Unit 1 Technical Specifications and Bases 2.1, 3.1, 3.4, 3.6, & 5.5.8

Technical Requirements Manual Revision 6, TRM 3.4 & 4.2.

Arkansas Nuclear One Unit 1 FSAR Section 1.4, 3.1.2.4.3, 3.2, 4.0, 5.1, 5.2, 6.3, 6.6, 7.2.2, 9.3.2.1, 9.6, 14.1, 14.2 and 16.2
Chapter 1Tables
Chapter 4 Tables
Chapter 9 Tables
Chapter 14 Tables
Chapter 3 Figures
Chapter 4 Figures
Chapter 5 Figures
Chapter 9 Figures

5. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☐ **Yes**
☒ **No**

If “YES,” list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: _____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10 CFR 50.59 Review Program Guidelines for guidance.

The functions of the Control Rod Drive Service Structure are:

- Seismic restraint for the CRDMs
- Seismic restraint for the RADCAL
- Provide lateral restraint for the CRDMs and RADCAL
- Provide support for the reactor vessel high point vent
- Provide radiation shielding (NPO blankets)
- Provide ventilation air flow path
- Provide support for CRDM work platform
- Provide support for the CRDM ventilation ductwork
- Provide support for RVCH tensioning and de-tensioning hoist and removal of studs
- Provide support for ancillaries required for the operation of the CRDMs
- Provide access to the CRDM work platform
- Provide access for attachment of the RVCH lifting rig assembly to the RVCH lifting lugs
- Provide support for the power and instrument cables (batwings).

The functions of the RVCH lifting rig assembly:

- In conjunction with the containment polar crane, lift and move the RVCH with CRDSS from the reactor vessel to the head storage stand
- In conjunction with the containment polar crane, lift and move the RVCH with CRDSS from the head storage stand to the reactor vessel

In conjunction with the containment polar crane, the lifting assembly is used to transport the RV internals.

The impact of the following modifications to the CRDSS design functions have been determined to be positive or neutral:

- The replacement CRDSS is comprised of upper and lower service structure cylinders in place of the existing single cylinder. This enhances the initial and future assembly, disassembly and installation of these components. The replacement configuration will have no adverse impact on the structural adequacy of the CRDSS.
- The design of the replacement CRDSS Batwings allows the Batwings to remain attached to the CRDSS platform while the RVCH and CRDSS assembly is moved. The Batwings will still provide the same functionality as the existing Batwings, but will decrease the time and effort required to position the Batwings for RVCH removal and reinstallation. This will reduce personnel radiation exposure and enhance safety. The replacement batwing configuration will have no adverse impact on the structural adequacy or functionality of the CRDSS.
- When fabricated, the existing CRDSS was classified as non-safety related and the ASME B&PV Code requirements were not imposed for the design or construction of the assembly. The replacement CRDSS is safety related except for the hardware above the work platform deck plates, platform handrails, ICW piping, ventilation ductwork, service air system, shielding, stud tensioner monorails and Batwings. These items are augmented quality as required by SPEC-ANO-M-579, “Design Specification for Replacement Control Rod Drive Service Structure – ANO-1”. This ensures at a minimum that the replacement CRDSS is structurally equivalent to the existing CRDSS.
- The three existing 2-ton air operated chain hoist will be replaced with three new 6600-lb (de-rated to 4400-lb with the use of the stainless steel load chain) lifting capacity air operated chain hoist. The replacement configuration has been analyzed with the results determining that the replacement stud tensioner handling hoist and trolley configuration when remaining in place during power operation would have no adverse impact on the structural adequacy or functionality of the CRDSS.
- The Intermediate Cooling Water (ICW) headers have been redesigned to facilitate ease of connection and disconnection of the tubing spool pieces (flex hoses) used for supplying the cooling water to the CRDMs. The redesigned headers will have fewer penetrations with each penetration servicing more CRDMs. New supply and return flex hoses will be installed. This reduces the quantity of ICW piping for better access to the tops of the CRDMs. To offset the expected increased pressure drop due to the increased flow in the branch headers in the new design, the inlet and outlet branch header pipes were increased in size from 1¼” SS tube to 1½” Sch, 40 SS. The new design was evaluated with the results showing that the CRDM cooling water manifolds and longer flexible hoses did not increase the pressure drop and are hydraulically acceptable for cooling the CRDMs. The original cooling water piping was non-safety and therefore detailed seismic analysis (category I seismic analysis) was not required. There was no reason to classify the piping as category I seismic then or now according to RG 1.29 or 10 CFR Part 50. Since II/I criteria was not in place formally at that time, seismic stability was not formally required so if any seismic was evaluated no requirements for documentation were in place. The new cooling water piping is also non-safety and as with the original piping it does not require seismic category I analysis, but was analyzed to seismic II/I criteria using ANO-1's prescribed techniques and plant specific requirements. The new ICW headers and supports were evaluated and determined to be capable of meeting all of the qualification and functional requirements of the existing CRDM cooling water configuration.
- There will be eight (8) 10 inch-by-12 inch ventilation holes in the lower section of the CRDSS to accept the connection of the redesigned ductwork. The existing CRDSS utilized eleven (11) 12-inch diameter and one (1) 12 inch-by-18 inch elliptical ventilation ports. Only eight (8) of these ports were used for supplying ventilation with the remaining four (4) ports covered during plant operation. The CRDSS with the ventilation ports has been evaluated with the results determining that the ventilation ports would have no negative impact on the structural adequacy of the CRDSS.
- An elliptical opening measuring 14 X 18-inches is located approximately 20 3/8 inches from the bottom mounting flange on the replacement CRDSS. This opening will provide personnel and equipment access to the interior CRDSS chamber and will be covered during plant operation. This opening was evaluated with the results determining that the opening would have no adverse impact on the structural adequacy of the CRDSS
- To enhance the ability for future inspections, inspection ports are located vertically on center at 8.88 inches from the lower cylinder’s bottom flange. These ports are considered a design enhancement and have been evaluated with the results determining that the inspection ports would have no adverse impact on the structural adequacy of the CRDSS.

- A full circumference band is provided to cover the elliptical openings and the smaller inspection ports when they are not being used. The cover was evaluated and determined to not present an adverse affect on the function of any SSC.
- The existing support structure for the radiation shielding blankets is attached by a banding configuration with the foot of the structure on the CRDSS lower flange. The support structure on the replacement CRDSS will be welded to the lower CRDSS cylinder assembly above the ventilation ductwork ports. This structure will support a monorail that the existing trolleys and shielding blankets would be attached to. The structural adequacy of this configuration was addressed. The functionality and capability of the existing radiation shielding will be unchanged by the replacement of the CRDSS.
- The existing RV vent line/High Point Vent (HPV) System will have the electrical connections disconnected and the mechanical portions will be discarded or salvage as appropriate. New piping materials, supports and solenoid valves will be erected to the replacement RVCH/CRDSS assembly prior to moving the assembly into containment. The RV vent line/HPV System will be connected to core location N6 the same as on the existing RVCH. The replacement RV vent line/HPV System will be installed using the existing drawings to maintain the existing configuration. A mechanical and electrical equivalency qualification was performed for the replacement valves used in the HPV System application. The HPV venting flow requirements and the effects of using the reduced Cv valves were addressed and evaluated as acceptable. Seismic and structural qualification of the new valves was performed and determined to be acceptable. PT-1070 and TE-1070 will be reinstalled in the new HPV piping so there will be no change in the vent or monitoring capability of the RV vent and HPV system.
- New service air manifolds will be attached to the replacement CRDSS work platform. These manifolds would be supplied with air during maintenance and inspection activities and provide local connections for equipment and tools. The air manifolds were evaluated with results that the manifolds and supports meet the seismic requirements and are acceptable for use with the replacement CRDSS. This is considered an enhancement in personnel safety and will provide a reduction in exposure time for all activities performed on the RVCH and CRDSS that require a supply of service air.
- The appropriate carbon steel sections of the replacement CRDSS will have Carboguard 890N coating which is approved for use inside containment. The coating was applied per ANO approved procedure. These controls will ensure that the appropriate protection of the carbon steel is provided and that there will be no adverse impact on the containment sump from the coating applied.
- When the FPAs are not attached to the lifting tripod, lateral movement is restrained by squeeze plates fastened to the slotted cover plates to lock the pendants in place. The slotted cover plates are attached to the platform and provide a path for the pendants to move during lifting operations. The squeeze plates were evaluated and determined to have no impact on the functionality of the CRDSS.

The following modifications will be performed to the RVCH lift rig assembly:

- The existing braided wire ropes would be replaced with fixed pendants assemblies (FPAs) that pass through similar openings in the platform floor plates and support structure of the CRDSS.
- The ends of the pendants are threaded with the lower end threaded into a clevis for connection to the RVCH lifting lugs and the upper threaded into a turnbuckle. The other end of the turnbuckle has a T-Lug threaded into it. All threaded connections have a lock nut to ensure connections will not unthread during use.
- The T-Lug is designed to be used with the lifting fixture latch boxes. This enables the tripod lift fixture to remain in the same configuration for lifting the RV internals and the RVCH. As a result the tripod can be stored with the lifting fixture latch boxes attached, which reduces handling time and possible handling accidents.
- The existing RVCH lift rig is only attached to the RVCH lifting lugs during plant shutdown activities that require RVCH movement. The replacement FPAs remain attached to the RVCH lifting lugs during all modes of plant operation.
- The existing internals handling adapters will be replaced with like internals handling adapters, which are rated for increased lift capacity. Interface and connection to the FPA T-Lugs and reactor vessel internals remains unchanged.

The new RVCH lift assembly was evaluated with results demonstrating that the lifting assembly is qualified to perform all the required lifting functions and support functions as well as meeting all ASME Code seismic requirements.

The Reactor Vessel Head Stand and the floor loading in the head stand area were evaluated for the additional weight and were determined to be adequate to support additional weight and therefore would not increase the likelihood of a malfunction or compromise the function or mitigating capabilities of any SSC.

A new maximum head lift height above the reactor vessel was established. This new maximum height still provides margin to clear the top of the Alignment Guide Studs while staying within the risk associated with carrying heavy loads over the open reactor vessel during refueling (such as the reactor vessel head). "Unit 1 Reactor Vessel Closure Head Removal and Storage", procedure number 1504.007 will be revised to incorporate the new maximum lift height.

In summary, the above changes do not involve an adverse change in Design Function.

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

Accidents and events evaluated in the FSAR that could be impacted by this proposed activity would be:

- Rod withdrawal at rated power
- Stuck-out, stuck-in, or dropped Control Rod Accident
- Rod Ejection Accident
- Loss of Coolant Accident.

The replacement CRDSS with appendages, modified RVCH lifting assembly and TSS have been evaluated and determined to meet or exceed all code and owner's requirements for fabrication, installation and testing. The results determined that all elements met the requirements and there would be no increase in frequency of occurrence of a failure of any equipment or components that could cause a transient and therefore no increase in the likelihood of initiating an accident.

This activity will not change how control rods are supported, restrained, powered, monitored or operated; therefore there will be no change in occurrence to any accident associated with the control rods or control system.

This activity will not reduce the current pressure boundary capability of any component and therefore would increase the likelihood of a primary system pressure boundary breach.

The new RVCH lifting assembly was evaluated for structural integrity and for being a permanent component of the CRDSS during plant operation. The results demonstrated that the new lifting assembly was qualified to perform all the required lifting functions and support functions as well as meeting all seismic requirements: therefore the likelihood of a reactor vessel head drop is not increased.

The impact of additional combined weight of the RVCH and Service Structure has been evaluated with results determining that the additional weight was acceptable for the RCS supports and did not impact the reactor internals.

ER-ANO-2002-0640-000, "ANO-1 Polar Crane Up-rate to 190 Tons", provides the qualification of the polar crane to lift and control the combined additional weight of the RVCH and Service Structure.

The new Tripod Storage Stand (TSS) would not be an initiator of any FSAR evaluated accident.

The change in the containment heat sink and free volume has been evaluated and determined to have a minimal impact on the margins used in the containment peak pressure analysis. The Design Basis Accident (DBA) peak pressure value will remain unchanged and would not decrease the margin between peak calculated pressure and the reactor building's design pressure.

All electrical and instrumentation will continue to function within the same design envelope as currently required. Therefore, the replacement CRDSS with appendages, modified RVCH lifting assembly and TSS would not introduce an increase in the frequency of occurrence of an accident previously evaluated in the FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The proposed activity could impact the reactor building, the reactor coolant system and/or the ability of the control rods to control the reactivity of the reactor. The replacement CRDSS with appendages, modified RVCH lifting assembly and TSS have been evaluated and determined to meet or exceed all code and owner's requirements for fabrication, installation and testing.

This activity will not change how control rods are supported, restrained, powered, monitored or operated; therefore there will be no change in occurrence to any malfunction associated with the control rods or control system.

The impact of additional combined weight of the RVCH and Service Structure has been evaluated with results determining that the additional weight was acceptable for the RCS supports and did not impact the reactor internals.

ER-ANO-2002-0640-000, "ANO-1 Polar Crane Up-rate to 190 Tons", provides the qualification of the polar crane to lift and control the combined additional weight of the RVCH and Service Structure.

A structural/stress analysis of the connection and bracing element welds along with the bolted joints was performed with results determining that all elements met the Code requirements.

The new RVCH lifting assembly was evaluated for structural integrity and for being a permanent component of the CRDSS during plant operation. The results demonstrated that the new lifting assembly was qualified to perform all the required lifting functions and support functions as well as meeting all seismic requirements: therefore the likelihood of a reactor vessel head drop is not increased.

All electrical and instrumentation will continue to function within the same design envelope as currently required.

The pressure boundary capability of the CRDMs will not change.

The change in the containment heat sink and free volume has been evaluated and determined to have a minimal impact on the margins used in the containment peak pressure analysis. The Design Basis Accident (DBA) peak pressure value will remain unchanged and would not decrease the margin between peak calculated pressure and the reactor building's design pressure.

Therefore, the replacement CRDSS, modified RVCH lifting assembly and TSS would not increase the likelihood of occurrence of a malfunction of any safety or non-safety SSC previously evaluated in the FSAR or create any new malfunctions.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

Accidents and events evaluated in the FSAR that could be impacted by this proposed activity would be:

- Rod withdrawal at rated power
- Stuck-out, stuck-in, or dropped Control Rod Accident
- Rod Ejection Accident
- Loss of Coolant Accident

This activity will not change how control rods are supported, restrained, powered, monitored or operated; therefore there will be no change in consequences of an event involving the control rods or control rod control system. The CRDM nozzles have been demonstrated to be equivalent or better than the existing nozzles.

No new radiological sources, processes or materials have been introduced. This activity will not significantly increase the radiological activity levels in any contaminated system or reduced the capability of any barrier limiting dose rates to onsite or offsite personnel. No new flow paths have been created.

The increase in combined weight of the RVCH and Service Structure has been evaluated for impact on the RCS supports and the reactor internals and determined that the impact would be minimal and would not compromise the RCS integrity. The pressure boundary capability of the CRDMs will not change.

The change in the containment heat sink and free volume has been evaluated and determined to have a minimal impact on the margins used in the containment peak pressure analysis. The Design Basis Accident (DBA) peak pressure value will remain unchanged and would not increase the challenge to the containment capabilities.

During postulated FSAR accidents, SSCs important to safety would be required to operate under the same conditions and perform the same design functions, and would not be called upon to operate under more limiting conditions or provide greater mitigation capacities.

A new maximum head lift height above the reactor vessel has been established. This new maximum height still provides margin to clear the top of the Alignment Guide Studs while staying within the risk associated with carrying heavy loads over the open reactor vessel during refueling (such as the reactor vessel head). “Unit 1 Reactor Vessel Closure Head Removal and Storage”, procedure number 1504.007 will be revised to incorporate the new maximum lift height. ER-ANO-2002-0640-000, “ANO-1 Polar Crane Up-rate to 190 Tons”, provides the qualification of the polar crane to lift and control the combined additional weight of the RVCH and Service Structure.

The above provides assurance that the replacement CRDSS, modified RVCH lifting assembly and TSS would not provide more than a minimal increase in the consequences of an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

This activity does not introduce any new radiological sources, processes or materials. This activity will not significantly increase the radiological activity levels in any contaminated system or reduce the capability of any barrier limiting dose rates to onsite or offsite personnel. No new flow paths have been created. The functionality and capability of all safety and non-safety related structures, systems and components have been maintained. There are no reductions to any structure, system or component qualifications. There would be no reduction in capability, redundancy or separation of any structure, system or component for containing, monitoring, detecting or providing mitigation equipment initiation. There will be no change to the mitigation capability of any SSC used to address any off-normal occurrence or transient therefore, the consequences of a malfunction of any safety or non-safety SSC previously evaluated in the FSAR would not increase.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The replacement CRDSS with appendages, modified RVCH lifting assembly and TSS have been evaluated and determined to meet or exceed all code and owner's requirements for fabrication, installation and testing. Extensive analyses and evaluations were performed to confirm and document that the requirements were met and meeting these requirements assures that the existing accident analyses remain valid and bounding for all affected SSCs. The methods of equipment handling and storage introduced by this activity would not be initiators of any new type of accidents because 1) the equipment handling is only during preparation for refueling activities, 2) reassembly of plant during post refueling activities and 3) during plant operations provisions have been provided to adequately secure all equipment in an approved storage location. Therefore this activity would not create the possibility for an accident of a different type than previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

Types of malfunctions:

- Fluid leaks
- Power failure
- Loss of position indication
- Cooling flow blockage
- Cross contamination
- Erroneous indications
- Support failure
- Lack of heat removal

The replacement CRDSS, modified RVCH lifting assembly and TSS have been evaluated and determined to meet or exceed all code and owner's requirements for fabrication, installation and testing. Maintaining strict adherence to the applicable code requirements and owner's specifications provides assurance that the capability and functionality of the replacement CRDSS and modified RVCH lifting assembly will meet or exceed the requirements for the existing components. The replacement components will provide the same functionality as the existing components except for storage of items that are only used during refueling evolutions. The replacement components and new components are not credited for accident mitigation. The methods of equipment handling and storage introduced by this activity would not create any new type of malfunctions. This assures that a different result than any previously evaluated in the FSAR is not created and that FSAR identified malfunctions would continue to be bounded by the existing safety analysis.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

The review identified the possible DBLFPBs that could be impacted by the replacement CRDSS, modified RVCH lifting assembly, TSS or the additional combined weight of the RVCH and Service Structure:

- DNBR/MCPR: None of the above listed items would have an impact on the RCS volume, flow rates, flow paths or primary flow capabilities therefore all events having MDNBR acceptance criterion will still meet that acceptance criterion.
- Primary Pressure: This activity will not change any primary system parameter or limit. There would be no reduction in margins to any limit and there would be no change to how any SSC is operated or controlled. The transient and accident mitigating capabilities of all SSC's will be maintained; therefore all events having a maximum primary pressure acceptance criterion will still meet that acceptance criterion.

- RCS Boundary Stresses: The replacement reactor vessel closure head has been evaluated and determined that the stresses associated with the RCS boundary were acceptable for normal, upset, or faulted conditions. All components have been verified to meet the code and owner's requirements as required by ASME B&PV Code, Section XI. The increase in combined weight of the RVCH and Service Structure has been evaluated for impact on the RCS supports and the reactor internals and determined that the impact would be minimal and would not compromise the RCS pressure boundary.
- Containment Design Pressure: The change in the containment heat sink and free volume has been evaluated and determined to have a minimal impact on the margins used in the containment peak pressure analysis. The Design Basis Accident (DBA) peak pressure value will remain unchanged and would not decrease the margin to the reactor building's design pressure.

Therefore no design basis limit for a fission product barrier as described in the FSAR will be exceeded or altered by the replacement CRDSS, modified RVCH lifting assembly, TSS or the additional combined weight of the RVCH and Service Structure.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

This proposed activity involves the installation of a replacement control rod drive service structure on a newly installed RVCH, the modification of the RVCH lifting assembly and a Tripod Storage Stand. The CRDSS, the modified RVCH lifting assembly and TSS have been evaluated to plant specific requirements, and relevant industry standards and guidelines. The industry standards and guidelines were used to ensure that replacement and modified components were qualified and approved for the appropriate application. This provided qualification and validation of the input from the affected SSCs to the FSAR evaluations, but no part of this proposed activity involved a change or revision to any method of evaluation described in the FSAR and used in establishing the design bases or in the safety analyses.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-020

I. OVERVIEW / SIGNATURES

Facility: ANO-1

Document Reviewed: ***ER-ANO-2000-2294-001, FW Heater E-4A/B & E-5A/B*** **Change/Rev.:** 0

System Designator(s)/Description: ***CS, EX, HD, HV***

Description of Proposed Change:

ER-ANO-2000-2294-001 replaces E-4A, E-4B and the bundles for E-5A & E-5B. The replaced E-4s are larger in diameter and the replaced E-5s are longer to accommodate a power uprate and maintain approximately the same hydraulic characteristics. This requires a shell extension on the west side of the E5 heater shells. The approximate 3 foot west-side extension requires relocation of piping and supports. Level controls are being reworked and each heater will utilize 2 sets of level taps. The Heater "HI" level alarm in the control room will become a "HI/LO" level alarm to protect against a loss of heater level and the subsequent drain cooler damage. In the process of performing our reviews, discrepancies were found in the Turbine Bldg structural steel calculations that require replacement or reinforcement of several Turbine building floor beams. Finally, some minor reinforcement of structural members inside the Unit 1 condensers will be required.

Screening of the proposed change against the description of the affected Systems, Structures, and Components (SSCs) in the SAR yields the following changes as described in the SAR:

1) ***(P 10.4.7) The new heaters are manufactured using the standards of the Heat Exchange Institute (HEI). This organization has replaced the Feedwater Heaters Manufacturers Association referenced in the SAR.***
2) ***(Figure 10-2) The feedwater heaters (E-4A/B & E-5A/B) are depicted as Straight-Tube heat exchangers. They are being revised to be shown as U-Tube heat exchangers.***
3) ***(Figure 10-2) The channel vent and drain connections HV-70, HV-72, HV-85, HV-94, HV-74, HV-83, HV-60, HV-62 and HV-3015 are being deleted from SAR figure 10-2 since the U-tube configuration removes the far side channel. This location will now be part of the shell and is more appropriately located on P&ID M-205 sheet 1, which is not a SAR figure.***
4) ***(Figure 10-2) To provide an additional temperature testing point downstream of the E4s and upstream of the bypass connection, TW-2811 and TW-2812 have been added.***
5) ***(Figure 10-2) Because of the configuration of the channel, additional channel vents are required to vent the space above the partition cover plate during start-up. As such, vent lines were added with the following globe isolation valves: HV-1036, HV-1037, HV-1034, & HV-1035.***
6) ***(Figure 10-2) Relief valves PSV-2914 & PSV-2908 will be shown as directly connected to the heater channel instead of the channel outlet nozzle.***
7) ***(Figure 10-2) The channel nozzles are being shown as either straight pipes (E5 outlets) or reducers (E4 inlets & outlets and E5 inlets) to depict the actual configuration of the new heaters.***
8) ***(Figures 1-4, 1-7, 1-9, & A-3) Equipment location drawings are being revised to show the full channel access covers and the footprint of the replacement heaters.***

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☒	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-020)**	**Sections I, II, and IV required**

Preparer: M. Harris / **ORIGINAL SIGNED BY MARK HARRIS** / EOI / Design Eng / 3-26-05
Name (print) / Signature / Company / Department / Date

Reviewer: D. Williams / **ORIGINAL SIGNED BY DAN WILLIAMS** / UPI / Safety Analysis / 3-26-05
Name (print) / Signature / Company / Department / Date

OSRC: J. R. Eichenberger / **ORIGINAL SIGNED BY J. R. EICHENBERGER** / 4-28-05
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings and 50.59 Evaluations.]

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM ENS-LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	1SAR10.4.7; SAR Figures 1-4, A-3, 1-9. 1-7 & 10-2
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES", perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. <u>AND</u> initiate an LBD change in accordance with NMM ENS-LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM ENS-LI-113. No further 50.59 review is required.			

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1 If "YES," see Section 5.2[5]. No LBD change is required.
2 If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Review.
3 Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
4 If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☐ Yes
☒ No

If "yes," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. Basis

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

Operating License/Technical Specifications:

A review of the operating license reveals that the proposed FW heater replacements do not impact any of the conditions or requirements specified by the NRC as a condition for operating ANO-1. Although the heat rate should improve, the steady state core thermal power is unaffected. During installation, compensatory measures will be taken by Security to assure NRC requirements for physical security are met. ANO Technical Specification section 3.7.3 include restrictions on the operation of certain main feedwater isolation and control valves; none of which are impacted by the proposed changes. A review of T. S. Design Features (Chapter 4.0) and Administrative Controls (Section 5.0) revealed no impact. Measures are being taken to assure that the Secondary Water Chemistry Program (Section 5.5.10) is not adversely impacted by the installation of the proposed change. These measures include active participation by the ANO Chemistry Department during fabrication of the FW heaters by sampling the vendor's demin water system and by testing chemicals used during fabrication.

FSAR:

FSAR Figures 1-4, A-3, 1-7, 1-9 and 10-2 (Items 2-8) are being revised to reflect the proposed changes. These figures are design drawings that have been incorporated into the FSAR. A minor change to the text in SAR Section 10.4.7 (Item 1) reflects that the Feedwater Heater Manufacturer's Association has be re-incorporated as the Heat Exchange Institute (HEI). Newer standards for FW heaters are issued by HEI. Item 1 is considered an editorial change only and therefore will not be discussed further in this evaluation.

In the FSAR description of a turbine overspeed event, the consequences are not evaluated based on an explicit or an implicit definition of the steam energy input from the LP FW heaters following a closure of the main turbine stop valves. The size of the LP FW heaters is therefore not relevant to the analysis of this accident. The consequences of the overspeed event are limited by the intervening concrete structures. The FSAR considers radiological consequences from a turbine overspeed event as not credible. It credits the plant turbine control and protection system, the high value of bursting overspeed, and the maintenance and inspections programs. Therefore the turbine overspeed accident in section 14.1.2.9 is not impacted.

Likewise for a main steam line break event, the consequences are not evaluated based on an explicit or an implicit definition of the steam energy input from the LP FW heaters. The HP FW heater impact on the MSLB analysis is discussed in section 14.2.2.1.3.4.4, however, this discussion is limited to the E-1 FW heaters only. As such the MSLB accident is not impacted.

Tests or Experiments considerations

Testing associated with the proposed changes includes pre-service tests to assure that the replacement components and their indications are functioning as designed and in-service tests to validate heater level set-points are adequate. None of the tests will force the plant to be operated in modes for which it is not already analyzed. Operation at low feedwater heater levels for a short period of time is necessary to properly establish a safe operating level; it will not cause significant damage to the replacement feedwater heaters. None of this testing constitutes a test or an experiment not described in the FSAR.

4. **References**

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.5.1[5](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
(ZyFind – 50.59 – Unit 1)	***("E4" OR "E-4" OR "E5" OR "E-5"; "3069" OR "3107")***
(Autonomy (ANO) – 50.59 – Unit 1)	

LBDs/Documents reviewed manually:

(1SAR Chapter 10)

5. **Is the validity of this Review dependent on any other change?**

☐ **Yes**
☒ **No**

If "YES", list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure ENS-EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure ENS-EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☒	☐	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☒	☐	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☒	☐	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

Replacement activities will require bringing 4 large FW heaters onsite (approximately 4.5 ft in diameter by 50 ft long). Any complete security inspection of their internals will have to be performed at TEi's facility in Tulsa before the shells are installed. Channel side inspections can be performed onsite when they arrive. Shipping will be by truck and will likely require security barrier removal at the plant's North entrance. Civil Design Engineering will evaluate the vehicle pop-up barriers. Once the heaters arrive they will be located just East of the radiography building near the rail spur. At this location NDE will be performed on the heater tube bundles. Just prior to the outage the heaters will be moved into the protected area through the Sally port and stored Southeast of the turbine building. During installation, siding will be removed from the East turbine building walls and large rigging components will be installed over the condenser pulling pit.

To house construction crews, 2 trailers will also be located within the protected area during the outage. These will be located 15 feet East of the U1 oily water separator.

Activities affecting security equipment and security operations have been reviewed and a walk down completed with ANO-Security and ANO-Field Engineering. Appropriate compensatory measures will be put in place for those activities that affect security.

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

(NOTE: This section is not applicable to Waterford 3 and may be removed from 50.59 Reviews performed for Waterford 3 proposed activities.)

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure ENS-LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: ____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions.

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

The only change being performed under this 50.59 is a change to the ANO-1 criticality analysis for the Holtec HI-STORM 100 -24 cask design. Only Question 8 is necessary to be answered, however, the other 7 questions are being discussed for completeness.

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   Chapter 14 accidents reviewed for potential impact by the proposed change are the turbine overspeed accident discussed in FSAR Section 14.1.2.9 and the main steam line line break analysis in 14.2.2.1.

   Turbine overspeed accident initiators implied in the discussion are failure of the turbine plant control system, contaminated control system fluid, material failure of the turbine disc or steam admission valve failure(s).

   Main steam line break accident initiators are not discussed in SAR Section 14.2.2.1. They are assumed to occur and the effects and consequences are evaluated. Changes 2) – 8) therefore cannot affect any accident initiators.

   The proposed changes to the E4 and E5 heater designs (items 2-8) do not impact these SAR accident initiators; therefore the proposed changes do not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   Heater replacement is part of ANO's site program to replace aged components that are subject to failure. Heater failure could cause an unnecessary secondary plant transient and a subsequent loss of generation capability but it would have not direct or indirect (EQ, steam impingement, flooding, etc.) impact on any important to safety SSC due to the location of the heaters. Heater replacement is desirable and justified from a business point of view. Because a heater malfunction would not have any adverse effect on any important to safety SSCs proposed changes 2) – 8) do not result in more than a minimal increase in the likelihood of an occurrence of a malfunction of an SSC.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   From the description of the screened changes above (items 2-8), there exists no information of a quantitative nature that was described as an input to the FSAR accident analysis. For instance, the FSAR credits the plant turbine control and protection system, the high value of bursting overspeed and the maintenance and inspections programs for the prevention of a main turbine overspeed accident. It does not credit the size or type of heaters. None of the proposed changes (Items 2-8) impact these items.

   The affected equipment is related directly to power production and equipment operation, monitoring and protection. The affected components are not used for accident mitigation.

   Since the proposed changes (items 2 – 8) will have no adverse impact on the consequences (dose received) it is concluded that they will not result in more than a minimal increase in the consequences of an accident.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The heaters are located in the condensate system upstream of the main feedwater pumps and the main feedwater isolation valves (MFIV). The pressure drop across the replacement E4s and E5s taken together will be approximately the same as the existing heaters. Even with a minor change it is difficult to postulate a scenario that would create greater consequences from a failure of the MFIVs. If they failed open the anticipated flow would be approximately the same. Therefore the proposed changes (items 2 – 8) will not result in more than minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The replacement heaters are designed and installed to codes and standards applicable to this type of component/system. The heater's function remains the same. New accident scenarios as a result of these proposed change not already evaluated in the FSAR are not credible. As such, the proposed changes (Items 2-8) do not create a possibility for an accident of a different type than any previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

None of the screened changes 2-8 affect components that are important to safety nor are there any important to safety components in close proximity to these changes. Therefore it is difficult to postulate any direct or indirect affect on an important to safety SSC. Additionally, malfunctions identified in Table 10-1 of the FSAR have no relationship to the proposed changes 1 - 8. The affected structure, systems and components (i.e. turbine building; EX, CS, & HD systems; heaters) are not evaluated in the FSAR for failure. Therefore the proposed changes do not create a possibility for a malfunction of an SSC important to safety with a different result than previously evaluated.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

Fission product barriers are identified as the fuel cladding, the RCS boundary and primary containment. The proposed changes on secondary plant systems have no impact on or interface with these design features and as such do not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

The method of evaluation of a missile generating event for a turbine overspeed is based on a review of anticipated failures at speeds above 1800 RPM. The evaluation compared the failures that could likely produce high energy missiles. The evaluation notes that failures of generator internals, blade rubbing and likely bearing failures create large breaking loads, implying that these loads would likely slow the rotor before high energy missiles from the LP turbine blades would be created.

Complex MSLB analysis methodologies evaluate the impacts on the RCS and the mitigating effects that the secondary plant response has on a MSLB accident. Subsequent evaluations looked at the impact that E-1 heater replacements had on the MSLB event. They conclude that the results were bounded by the previously performed analysis.

The proposed changes (Items 2-8) do not affect a method of evaluation and therefore do not constitute a departure from SAR methods of evaluation of these accidents.

If any of the above questions is checked "YES", obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure ENS-LI-113.

ANO 50.59 Evaluation Number

2005-022

I. OVERVIEW / SIGNATURES

Facility: **ANO - Unit 1**

Document Reviewed: ER-ANO-2002-0640-000, Polar Crane Up-Rate to 190 Tons
Change/Rev.: ERCN 4

System Designator(s)/Description: Polar Crane (L2)

Description of Proposed Change

The Unit 1 Reactor Building Polar Crane is an electric, circular traveling bridge crane with a single trolley equipped with both main and auxiliary hoists. It is utilized during unit outages for maintenance tasks such as removal/replacement of the reactor vessel closure head (RVCH) and upper plenum and other outage activities. The capacity of the Polar Crane main hoist will be up-rated from 150 tons to 190 tons. The auxiliary hoist (25 ton capacity) will not be up-rated. Additionally upon the implementation of this ER, the Polar Crane will be qualified for the following conditions:

1. It is qualified to Seismic Class I criteria when unloaded in its parked position.
2. It is qualified to Seismic Class I criteria when unloaded with the trolley and bridge in any position.
3. It is qualified to Seismic Class I criteria with a load of up to 10,000 lb on the main or auxiliary hook. This condition primarily applies to plant modes 3 and 4. If loads must be taken over required safety related components, a drop analyses shall be performed by Design Engineering to meet ANO's commitments to NUREG-0612.
4. It is qualified to Seismic Class II/I criteria with loads up to 190 tons in any position in Modes 5, 6 and defueled; however, ANO's commitments to NUREG-0612 shall be met.

The Polar Crane is being up-rated in order to have the capacity to lift the replacement RVCH (RRVCH) and replacement Service Structure (RSS) along with radiation shielding/stud bolts as a non-engineered lift in accordance with ANSI B30.2-1996. However, it should be noted that the Polar Crane will remain a non-single failure proof crane.

While the primary focus of the modification is to up-rate the main hoist, additional work (e.g., replace bridge drive motors and bridge wheels) is also being performed to further improve the performance of the Polar Crane. The following work is to be performed as part of this ER:

Main Hoist Wire Rope – The present 1.25" diameter IPS (improved plow steel) 6x37 classification wire rope will be replaced with 1.25" diameter XXIP (extra extra improved plow steel) 6x37 classification wire rope with a breaking strength of 87.9 tons. The reeving will remain 12-part wire rope.

Main Hoist Bottom Block Replacement (including the hook, power rotator and control cable) – A complete new bottom block assembly will be installed including a new hook, sheaves, clevis, and power rotators. The power rotators are designed to eliminate oil leakage. The dimensions of the new hook remain the same as the existing hook but stronger materials will be used for the higher ratings of the clevis, hook, and pins. Replacement electrical cable for the power rotators will be included.

Main Hoist Gearbox – All of the gears and shafts will be replaced from the end of the Magnetorque brakes to the drum gear including bearings, retainers, seals, and keys. All of the gears will be hard surfaced with the exception of the motor gear. The gearbox will be cleaned and all old grease will be replaced with new grease.

Main Hoist Drum and Gear – The drum will be replaced with a new drum including a new surface hardened drum gear and new high strength drum key. The replacement drum will have the wire rope preinstalled.

Main Hoist Drum Supports – The main hoist drum pedestals will be reinforced to support the increased capacity of the main hoist and seismic loads. This involves welding additional structural steel to the pedestal supports.

Trolley Wheels – In order to facilitate the increase in capacity of the main hoist without having to strengthen the existing bridge girders, the four wheels on the trolley will be replaced with a six-wheel design. These assemblies necessitate an increase in the height of the overall trolley assembly by 30". The drive wheels on the auxiliary hoist side of the trolley will be replaced with raised steel adapter supports and smaller diameter wheels. The main hoist side will be replaced with double wheel assemblies with equalizing bogies. New trolley gear reducer and axles are included in the modification. The existing motor will be reused.

Trolley Drive – The trolley drive will be changed by relocating the existing drive motor and brake from the top of the trolley to the new trolley drive assembly on the opposite side of the trolley. The power supply wiring for the trolley drive components will also be relocated/replaced.

Trolley Lateral Seismic Restraints – The restraints will be replaced with extended restraints due to increased height of trolley assembly.

Bridge Wheels – The eight bridge wheels and bearings will be replaced with "prescription wheels" and bearings to eliminate rubbing between the wheel flanges and the inside of the crane runway rail. Like the existing wheels, the prescription wheels are double flanged wheels. However, the prescription wheels have a specifically designed offset from the centerline of the crane rail to guide from the outside of the rail.

Bridge Drive Motors – The two existing 10 HP bridge drive motors will be replaced with 15 HP bridge drive motors. The bridge motor resistors and reactors will be replaced with those designed for a 15 HP motor. New motors will be the same frame size and type as existing bridge drive motor and will be compatible with the existing electrical equipment and configuration. The new bridge motors will incorporate integral disc brakes which will replace the existing bridge shoe-type brakes.

Electrical Pickup Cables on Trolley – Due to the increase in height of the trolley assembly, the cable length must be longer. Replacement electrical pickup cables for the existing pickup cables will be installed. Hypalon jacketing will be used.

Electrical Collector Bar – Collector bar extensions will be installed to extend the collector bar from the trolley to the existing electrical busses on the bridge and to accommodate the modifications of the trolley being raised in height.

Bridge Capacity Nameplate – The bridge capacity plates will be replaced to reflect the increase in main hoist capacity to 190 Tons.

Load Pins – Two new bottom block clevis pins will be installed to transfer load from the hook and lift fixtures to the bottom block. These pins are identical and interchangeable. The new pins are designed to support the new main hoist rated load of 190 tons.

Special Use Platform – A special use platform will be utilized to service the crane trolley and facilitate installation of the upgrade materials. This platform will provide lifting devices and personnel access required to support modification activities. Since some activities will be performed during the refuel mode additional restraints are included to prevent the platform from becoming dislodged from the polar crane and falling to the floor below. Also any heavy loose components will be additionally restrained as necessary to ensure that they also will not fall to the floor area below. This temporary platform will be removed upon completion of the activities described in the ER. To provide an alternative to the use of additional restraints, a qualitative evaluation of the maintenance risks associated with relatively short-duration configurations will be performed under 10CFR50.65 and documented in the ER. Additional variations in the plan are allowed where approved by appropriate Operations personnel with knowledge of the Shutdown Operations Protection Plan (SOPP), subject to the condition that the changes are reviewed under 10CFR50.65 and the review documented.

Ladders – Ladders will be installed, replaced or modified as required to provide safe access to crane components. A new trolley access ladder will be added to the B bridge girder for access from the trolley to the B girder access platform. Modifications and extensions to existing ladders will be performed in order to provide safe access with the increase of trolley height above the bridge

Bridge End Tie Bolts – New replacement bridge end tie bolts will be installed. These bolts will allow movement during seismic events. The bolts will be wired such that they will not fall from the crane should the bolts shear during a seismic event.

Additionally as part of this ER, the design safety classification of the Polar Crane in the ANO Component Database will be changed from "N" (non-safety related) to "S" (Special). ER-ANO-2002-0640-001, "Design Requirements for Unit 1 Polar Crane (L2) Upgrade," provides technical justification for changing the safety classification to "S" for seismic requirements and structural integrity. ER-ANO-2002-0640-001 provides quality requirements for all structural load retaining components. All electrical components are to remain classified as non-safety related. All structural and seismic calculations will remain "safety related".

Plant Modes/ Implementation Restrictions

No work on the Polar Crane for the up-rate is scheduled to be performed during Modes 1 – 4. As determined by construction, activities such as pre-fabrication, preparation, and pre-staging of equipment and materials outside the Reactor Building may be performed at any time before or during the OTSG/RVCH replacement outage.

In Modes 5 and 6, the Polar Crane will be used for defueling operations at the beginning of the 1R19 outage prior to any work on the Polar Crane up-rate. Crane inspections and staging of equipment may take place during these modes. Typically the crane is inspected prior to performing any lifts during an outage per plant procedures.

Work associated with the removal and installation of temporary components (such as access platform, blocks, and jacks) and permanent components under this ER will be performed while the Unit 1 reactor is in Modes 6 or defueled. The majority of heavy load movement will be performed when the reactor is defueled, however some activities will be performed during refuel activities. All conditions associated with either mode is appropriately identified in the design package. Some of these activities are evaluated under 10CFR50.65 instead of 10CFR50.59.

Inspections and testing may be after entry into Mode 6 (Refueling) or near the close of the 1R19 outage.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-022, Rev. 3)**	**Sections I, II, and IV required**

Preparer: H. C. Chadbourn/ **ORIGINAL SIGNED BY H. C. CHADBOURN** /EOI /Design Engineering/ 12-01-2005
Name (print) / Signature / Company / Department / Date

Reviewer: Steve Bennett/ **ORIGINAL SIGNED BY STEVE BENNETT** / EOI / LIC / 12-01-05
Name (print) / Signature / Company / Department / Date

OSRC: J. R. Eichenberger / **ORIGINAL SIGNED BY J. R. EICHENBERGER** / 12-01-05
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	Sections 5.1.2.1.2, 5.1.7.1, 9.6.1.6, and 9.6.1.7; Figures 1-7 and 1-11
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES", perform an Exemption Review per Section III OR perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☐ Yes
☒ No

If "yes," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. Basis

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.3.2 for guidance.

Operating License/Technical Specifications (TSs)

The Polar Crane is not discussed in the Operating License or Technical Specifications. Likewise, the Polar Crane is not discussed in the Tech Spec Bases, Technical Requirements Manual, or Core Operating Limits Report. Therefore, no changes to these documents are required nor are the documents impacted.

FSAR

The Polar Crane is discussed in a number of Sections in the FSAR. The Polar Crane is discussed as a Seismic Category 1 system in Sections 5.1.2.1.2 and 9.6.1.6, and these sections will be revised to reflect the new qualified configuration of the Polar Crane. Section 5.3.1 also mentions that the Polar Crane is a Seismic Category 1 system but the level of detail of this section is not affected by this ER. In Section 5.1.7.1, it is noted that the Polar Crane support structure was based on a crane hook load of 180 tons. This load will be revised to 190 tons. Section 9.6.1.7 notes that the Reactor Building crane (Polar Crane) has a rated load capacity of 150 tons. This will be changed to 190 tons. In addition, Section 9.6.1.7 will be revised to reflect that while the original welding was done in accordance with AWS D2.0 the Polar Crane up-rate welding is performed in accordance with AWS D1.1 which superseded AWS D2.0. Figures 1-7 and 1-11 will be revised to indicate the new hook heights and trolley wheel configurations. Figure 1-2 also shows the Polar Crane but no revision is required to this drawing. Other FSAR sections (e.g., 9.6.2.2) describe the uses of the Polar Crane and will not be revised since the level of detail is not impacted by this ER.

Test or Experiments Consideration

The proposed upgrade of the Polar Crane will increase the load handling capability of the crane. It will not result in the Polar Crane being used for functions that are outside of its design basis. Following modification installation, the crane will be inspected and load tested to 125% (+0%, -5%) of the rated load to verify proper installation of components and code/procedure compliance. Upgrade of the Polar Crane is not considered a test or experiment as defined in ANO-1 LBD's.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LRS 50.59 – Unit 1	polar w/10 crane*, reactor w/10 crane*, "L-2", "L2", NUREG*0612, "wire rope", "wire ropes", hoist*, trolley*, bridge*, "150 tons", "150 ton", "head drop", coating*, uncoated, "lift height", Magnetorque, "load path", "load paths"

LBDs/Documents reviewed manually:

ANO Unit 1 FSAR	Sections 5.1.4, 5.1.7, 5.1.8, 5.3.1, and 9.6.1.7 Figures 1-2, 1-7, and 1-11

5. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☐ Yes

☒ No

If "YES", list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1, above.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section II.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: _____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10 CFR 50.59 Review Program Guidelines for guidance.

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

Most accidents evaluated in ANO-1 FSAR Chapter 14 occur while the unit is at power. The only potentially applicable accident is the Fuel Handling Accident which occurs in the spent fuel pool in the Auxiliary Building or in the Reactor Building while the unit is in Mode 6 – Defueling/Refueling. With the exception of the load test, lifting of the lower load block, and removal of the work platform activities associated with major load handling (i.e. drum, wheels-except for the outer bridge wheels, etc), in the Reactor Building will occur while the reactor is defueled. For activities performed during refueling, the design package requires either that the work platform be restrained, in addition to the existing support arrangement, for seismic concerns, or that additional controls be in place that reduce risks to an acceptable level. The evaluation of configurations that are not seismically qualified or provided with redundant restraints to prevent failure due to seismic is performed in accordance with 10CFR50.65 and documented in the ER. The platform location during this mode will not be over components required to maintain safe shutdown and any additional risks are minimized. Load testing and refueling operations will be controlled in such a way that no loads will pass over fuel or required safety related equipment. Furthermore, a Fuel Handling Accident is initiated by the drop of a fuel assembly. The Polar Crane is not used to transport fuel assemblies. Therefore the implementation activities of this ER will not increase the frequency of a Fuel Handling Accident. The Polar Crane is not an accident initiator to any of the accidents described in Chapter 14 of the FSAR.

Therefore, it is concluded that the changes installed by ER-ANO-2002-0640-000 would not increase the frequency of an accident previously evaluated in the FSAR, during implementation of or as a result of the ER.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The Polar Crane is classified as "S". The reason for this is the Polar Crane's seismic qualification. At the completion of ER-ANO-2002-0640-000, the Polar Crane will be qualified for the following conditions:

- It is qualified to Seismic Class I criteria when unloaded in its parked position.
- It is qualified to Seismic Class I criteria when unloaded with the trolley and bridge in any position.
- It is qualified to Seismic Class I criteria with a load of up to 10,000 lb on the main or auxiliary hook. This condition primarily applies to plant modes 3 and 4. If loads must be taken over required safety related components, a drop analyses shall be performed by Design Engineering to meet ANO's commitments to NUREG-0612.
- It is qualified to Seismic Class II/I criteria with loads up to 190 tons in any position in Modes 5, 6 and defueled; however, ANO's commitments to NUREG-0612 shall be met.

These design requirements in addition to the Polar Crane up-rate from 150 tons to 190 tons demonstrate that the reliability of the Polar Crane to perform its design function and not become a Seismic II/I hazard has increased. In order to ensure that the Polar Crane is capable supporting the up-rated load, a load test will be performed to 125% (+0%, -5%) of the rated load. The load test will follow established load paths that have been evaluated for a load drop of the RVCH. The additional weight of the load test will not change this evaluation.

The modification affects the capability of the Polar Crane in future activities such as the RVCH and the indexing fixture lifts prior to and after fuel movements. A number of components that are a part of the Polar Crane will be modified or replaced in order to up-rate the Polar Crane from 150 tons to 190 tons per the requirements of applicable codes and procedures. The Polar Crane performs no safety related function; however, the Polar Crane is qualified as a Seismic Category I component in the parked position and in any position while supporting up to 10,000 lb from its main or auxiliary hook. The Polar Crane is also qualified as a Category Seismic II/I component while full loaded or unloaded during outages. The function and use of the Polar Crane will not change as a result of this ER. This ER will not degrade the Polar Crane but will up-rate the capability and reliability of the crane.

This ER does not change ANO's commitments to NUREG-0612 nor its requirements to perform load drop analyses. Additionally, operation and design of the Polar Crane will continue to remain in compliance with ANSI B30.2.

Therefore, the activities involved in ER-ANO-2002-0640-000 to up-grade the Polar Crane from 150 tons to 190 tons do not change the operation or function of the Polar Crane or increase the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

As previously discussed, the only potentially applicable accident during installation of this ER is the Fuel Handling Accident. Since some activities will be performed during refueling the possibility of components falling into the fuel pool are eliminated by incorporating special handling mechanisms within the guidance documents. No work will be performed directly above the transfer pool. Therefore components which could fall would not have a direct path to the pool area. Load testing will occur while the reactor is refueled or partially refueled; however, refueling and load testing operations will be coordinated so that Seismic II/I conditions will not occur during the load testing thereby eliminating the possibility of a fuel assembly drop. As previously noted, some of the configurations that will exist for short durations are evaluated under 10CFR50.65, and as such are outside the scope of this 10CFR50.59 Evaluation.

The only other potential dose increase that could occur after the completion of the activities described in this ER is the creation of a Seismic II/I hazard. As previously discussed, the Polar Crane has been qualified to meet either the requirements of Seismic Class I or Seismic II/I requirements. Therefore, the dose consequences of an FSAR seismic event are not increased. Additionally, this ER will not change the way the plant handles heavy load operations in relationship to ANO's commitments to NUREG-0612. Adherence to safe load paths will remain a plant requirement. The Polar Crane is not credited for mitigating any of the accidents described in Chapter 14 of the FSAR.

This modification does not introduce any materials that would result in hydrogen generation nor adds a significant amount of combustibles to the Reactor Building.

The up-rate of the crane to 190 tons will add approximately 11,000 pounds to the weight of the crane. This addition will be added to ANO Calculations 90-E-0060-01, "Containment Net Free Internal Volume" and 94-E-0043-01, "ANO-1 Reactor Building Surface Areas for the Heat Sink and Hydrogen Generation Calcs" in accordance with Engineering Report No. 94-R-1018-01, "ANO-1 Reactor Building Volume and Internal Surface Area Criteria". The addition of heat sinks is conservative and the reduction in net free volume (~22.4 ft^3) is within the current margin assumed in the Chapter 14 analyses. (Note that the addition of the above information to the referenced calculations is being tracked by a holdpoint in the ER.)

Therefore, the implementation or results of these activities do not alter the consequences (increase radiological dose) of a fuel handling accident (unit off-line) or other accidents (unit at power) described in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

In the event of malfunction of the Polar Crane, the consequences of a load drop are limited by the use of safe load paths and other programmatic procedures. The modification will not change the way that the plant uses safe load paths nor will it change the consequences of a load drop malfunction. In fact, this modification improves the reliability of the Polar Crane.

Therefore, any consequences as a result of a possible malfunction of the Polar Crane or other structure, system, or component (SSC) important to safety previously evaluated in the FSAR, will not be increased by the activities described in this ER.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

This ER involves the installation of a number of upgrades to the Polar Crane. These upgrades will be installed after the unit has been defueled or in the refuel mode. The Polar Crane will be load tested to 125% (+0%, -5%) of the new 190 ton rating prior to use. The load test will follow established load paths that have been evaluated for a load drop of the RVCH. The additional weight of the load test will not change this evaluation. As previously discussed, the seismic qualification of the Polar Crane in all conditions will be improved by the planned upgrades. This ER will not create any new load drop events. Therefore, the activities described in this ER will not create the possibility of an accident of a different type than previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The basic functions of the crane are to safely hoist, transfer, and lower loads as required by the plant. The crane is not designed to be “single failure proof” crane but it is equipped with redundant hoist brakes, seismic restraints and other items to minimize the possibility of dropping a load. The crane rating before this ER was 150 tons. At completion of this ER, the crane will be up-rated to 190 tons.

As previously discussed, this ER does <u>not</u> change ANO’s commitments to NUREG-0612 nor the requirements to perform load drop analyses. Operation and design of the Polar Crane remains in compliance with ANSI B30.2.

Appropriate safeguards have been provided to ensure maintenance activities will not create an unacceptable risk to systems required for refuel mode. The function of the Polar Crane will remain unchanged and the seismic capability will be upgraded as previously discussed. Therefore, this ER would not have created the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

ANO-1 fission product barriers include fuel cladding, RCS Boundary, and Containment Pressure.

<u>Fuel Cladding</u>
The planned upgrades to the Polar Crane will not begin until the core has been defueled. Upgrade activities are expected to be completed prior to full core reload, however some activities will occur during reload as identified in the design package. A number of the planned upgrades are performed on the Polar Crane main hoist and trolley. These activities could be a concern for loose parts entering the reactor if the work was performed over the reactor vessel. Before these activities begin, the Polar Crane trolley will be parked in a location away from

the reactor vessel. A special use platform will be utilized to facilitate installation of the upgrade materials and will serve to minimize fallen objects. Additionally, controls have been established for that portion of the modification performed during refuel activities to ensure components on the work platform, and the work platform itself, do not create an unacceptable level of risk during the short duration evolutions or configurations that are evaluated under 10CFR50.65. These controls include positioning of the trolley and work platform to minimize possible loose parts entering into the transfer pool.

Qualified coatings will be applied to new Polar Crane components in accordance with ANO procedures and the wheels and bearing housing will be installed uncoated. This will minimize the potential for coatings to come off in the event of an accident and cause sump blockage. The proposed Polar Crane upgrades will not result in a design basis limit for the fuel cladding barrier as described in the FSAR being exceeded or altered.

RCS Boundary
Most of the activities associated with ER-ANO-2002-0640-000 will be performed with the unit in defueled or in a refuel mode. Major load activities during refueling will involve the reaving of the main block. However adequate measures have been included in the design package to address maintaining components on the work platform. Work activities will be performed in an area away from the transfer pool which will minimize impact on the RCS boundary. Load testing may occur during or after refueling operations. At no time during the load testing will a Seismic II/I conditions exist over required SSCs. Therefore, the RCS pressure boundary will not be adversely affected by the proposed upgrades to the Polar Crane.

Containment Pressure
The Polar Crane girders and Polar Crane support structure were designed for 600 tons. The upgrade of the Polar Crane trolley to 190 tons does not affect the 600 ton rating of the Polar Crane support structure which is integral with the liner plate; therefore, the liner plate will not be adversely affected by the activities described in this ER.

The up-rate of the crane to 190 tons will add approximately 11,000 pounds to the weight of the crane which could negatively affect containment pressure calculations. The additional mass and surface area will be added to ANO Calculations 90-E-0060-01, “Containment Net Free Internal Volume” and 94-E-0043-01, “ANO-1 Reactor Building Surface Areas for the Heat Sink and Hydrogen Generation Calcs”, respectively, in accordance with Engineering Report No. 94-R-1018-01, “ANO-1 Reactor Building Volume and Internal Surface Area Criteria”. The addition of heat sinks is conservative and the reduction in net free volume (~22.4 ft^3) is within the current margin assumed in the Chapter 14 analyses. (Note that the addition of the above information to the referenced calculations is being tracked by a holdpoint in the ER.)

This modification does not introduce any materials that would result in hydrogen generation nor adds a significant amount of combustibles to the Reactor Building.

Therefore, the upgrades to the Polar Crane will not have an adverse impact on the Reactor Building as a fission product barrier.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

ER-ANO-2002-0640-000 modifies or replaces components of the Polar Crane in order to increase the rating of the Polar Crane from 150 tons to 190 tons. The Polar Crane, its design codes, and usage are described in Sections 5.1 and 9.6 of the FSAR. As noted in FSAR Section 5.3.1, the Polar Crane is a Seismic Category I structure. The method of analysis is not discussed in the FSAR. Therefore, this ER does not represent a departure from a method of evaluation described in the FSAR.

If any of the above questions is checked “YES”, obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure ENS-LI-113.

ANO 50.59 Evaluation Number

2005-023

I. OVERVIEW / SIGNATURES

Facility: ANO Unit 1

Document Reviewed: CALC-ANO-ER-05-030 (Part 50 Analysis of an MPC-24 for Unit 1) **Change/Rev.:** 0

System Designator(s)/Description: DFS

Description of Proposed Change:

This calculation documents the criticality analyses performed to satisfy the fuel loading activity at ANO Unit 1 specifically for the HI-STORM 100 24 assembly cask system in order to comply with the fuel loading requirements presented in the Unit 1 licensing basis for the Spent Fuel Pool (SFP).

The NRC issued Regulatory Information Summary (RIS) 2005-05, *Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations,* on March 23, 2005 which described a concern that when loading and unloading a dry cask in the SFP, the requirements of 10 CFR 50.68, *Criticality accident requirements*, should be met. The RIS notes that the requirements associated with preventing SFP criticality are included in General Design Criteria (GDC) 62, *Prevention of Criticality in Fuel Storage and Handling* and 10 CFR 50.68, *Criticality Accident Requirements*. Requirements associated with detection of SFP criticality events are described in 10 CFR 70.24, *Criticality Accident Requirements.* The RIS also highlighted the differences in the NRC Part 50 criticality requirements for the SFP and the Part 72 requirements for the spent fuel storage casks and emphasized that licensees are expected to comply with both Part 50 and Part 72 during cask loading and unloading operations.

The HI-STORM 100 Cask System Licensing Basis Documents already have criticality control requirements during loading and unloading operations of a spent fuel cask as required by Part 72; however these requirements are not the same as in Part 50. There are no Part 50 requirements specified in the Part 50 Licensing Basis Documents for cask loading. The RIS indicates that a cask system should be treated as a SFP rack from a criticality requirements perspective because there are no specific cask SFP criticality requirements. Based on the concerns raised by the NRC and reflected in RIS-2005-05, the criticality requirements for fuel assemblies in the SFP will be applied to the cask when loaded with fuel and in the cask loading pit while connected to the pool.

ANO Unit 1 currently is exempt from the requirements of 10 CFR 70.24 in accordance with the seven criticality requirements listed in the NRC's Information Notice 97-77. The Unit 1 SFP criticality analysis requirements are the same or more restrictive than the requirements outlined in 10 CFR 50.68 and from that perspective the analysis is bounding.

The analysis was performed in accordance with the SFP criticality licensing basis and it was shown that the current Unit 1 TS does not need to be revised or modified. In keeping with the detail related to storing fuel in the SFP racks, the TS bases, the Unit 1 FSAR, and the Unit 1 TRM will be revised for additional details related to the performance of the analysis for the cask.

Check the applicable review(s): **(Only the sections indicated must be included in the Review.)**

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-023)**	**Sections I, II, and IV required**

Preparer: Christopher Walker / **ORIGINAL SIGNED BY CHRIS WALKER** / EOI / DFS / 6-14-05
Name (print) / Signature / Company / Department / Date

Reviewer: Darrell Williams / **ORIGINAL SIGNED BY DARRELL WILLIAMSL** / UPI / DFS / 6-14-05
Name (print) / Signature / Company / Department / Date

OSRC: J. R. Eichenberger / **ORIGINAL SIGNED BY J. R. EICHENBERGER** / 6-16-05
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings and 50.59 Evaluations.]

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM ENS-LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	Section 9.6
TS Bases	☒	☐	Section 3.7.14, 3.7.15
Technical Requirements Manual	☒	☐	Section 3.7.8
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES", perform an Exemption Review per Section III OR perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM ENS-LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM ENS-LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☐ **Yes**
☒ **No**

If "yes," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

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3. Basis

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

Operating License:

The Unit 1 Technical Specifications (TS) have requirements for storing fuel in the spent fuel pools as follows:

- TS 3.7.14, Spent Fuel Pool Boron Concentration,
- TS 3.7.15, Spent Fuel Pool Storage, and
- TS 4.3.1.1, Criticality.

When the Unit 1 cask pit is open to the spent fuel pool (SFP) these TSs apply to fuel stored in the cask. The analysis performed addressed the Unit 1 TS bases and FSAR descriptions related to these TSs. All the requirements for storage in the spent fuel pool and accidents scenarios were performed for the HISTORM 100 24 assembly cask and it was shown that the minimum boron concentration requirement would not exceed the boron concentration required in TS 3.7.14 for the storage of fuel in the SFP (the TS limit of 1600 ppm).

TS 3.7.15 describe the combination of initial enrichment and burnup for fuel assemblies to be stored in the Region 2 SFP racks. This TS does not describe the loading requirements for the Region 1 racks, the upender, the new fuel elevator, or any other fuel storage location loading requirements. The analysis performed for the HI-STORM 100 24 assembly cask has shown that there are no burnup or enrichment limits that have to be imposed to store fuel within the cask.

TS 4.3.1.1 list the critical design parameters that shall be maintained in the spent fuel pool racks. The HISTORM 100 24 assembly cask criticality requirements are bounded by TS 4.3.1.1. TS 4.3.1.1 a. states that the maximum enrichment within the spent fuel pool is 4.1 wt%. Fuel assemblies must be stored in the SFP prior to loading in a cask, as such the maximum enrichment allowed in a cask on Unit 1 is 4.1 wt%. TS 4.3.1.1 b. requires k_{eff} to be less than or equal to 0.95 when fully flooded with unborated water considering the uncertainties listed in the FSAR. The analysis was performed with no credit for the SFP borated water and allowances for the uncertainties for the Region 1 racks were taken into consideration. The uncertainties described for the Region 1 racks were used because these racks most closely mimic the HI-STORM 100 cask with the use of a neutron absorbing material within the flux traps. TS 4.3.1.1 c. states the spent fuel pool rack nominal pitch where the HI-STORM 100 24 assembly cask has a slightly larger average nominal pitch. The ANO HI-STORM 100 24 assembly cask has a varying pitch throughout the canister as such the HI-STORM 100 24 assembly cask was modeled explicitly to account for all geometry effects. The controlling variable in this cell design is the flux trap spacing where a minimum flux trap dimension is maintained within the HI-STORM 100 CoC Appendix B, Section 3.2. Although the average center to center distance between fuel assemblies in the HI-STORM 100 24 assembly cask is larger than the center to center distance between fuel assemblies in the fuel racks, the larger distance results in a smaller k_{eff} for the fuel loaded in the HI-STORM 100 24 assembly cask.

Table 1 shows the relationship between the Part 50 and Part 72 fuel enrichment allowances and analytical assumptions, as well as the center to center distance between fuel assemblies placed in either the SFP racks or in the MPC-24. It is clearly demonstrated that the requirements of ANO-1 TS 4.3.1.1 are bounding.

Analysis	Fuel Enrichment (wt%)	Uncertainties Considered	Center to Center Distance between Fuel Assemblies in Storage Racks	Burnup Credit Required
ANO-1 TS 4.3.1.1	4.1	All	Nominal 10.65	Burnup
MPC-24 Part 50 Analysis Assumption	4.4	All	10.906 (>1.09 Flux Trap)	No Burnup
MPC-24 Part 72	5.0	All	10.906 (>1.09 Flux Trap)	No Burnup

Note: TS 4.3.1.1.d & e reference TS Figure 3.7.15-1 which provides fuel storage requirements for Region 2 based on a combination of initial assembly average enrichment and assembly average burnup. The MPC-24 Part 50 Analysis results concluded that there are no loading restrictions.

In keeping with the level of detail within the Unit 1 TS there are no additional spent fuel storage requirements that need to be added to the Unit 1 TS for storage of fuel within a HI-STORM 100 24 assembly cask. The parameters important to criticality that could warrant a TS requirement are bounded by TS 4.3.1.1. The operational requirements for the operation of the cask such as thermal, criticality, and radiological limits are specified within the HI-STORM 100 CoC (equivalent to a Part 50 TS) and can not be changed or modified without prior NRC approval and do not need to be changed to support this review. The Unit 1 technical specifications, operating license, and any NRC orders have not been impacted by these calculations.

Other LBDS:

The Unit 1 FSAR discusses the HI-STORM 100 dry fuel storage system and facilities in multiple places. Specifically, in the general fuel handling section of the FSAR, it is discussed that the dry fuel storage system is utilized to store fuel. The Unit 1 FSAR also states that the full descriptions for loading the cask are located in other documents. The site 10 CFR 72.212 report is referenced as one of these locations. However, in keeping with the premise that the cask should be treated as a spent fuel pool rack, the review will treat the cask from the stand point of the criticality review as the Unit 1 SFP Region 1 and Region 2 racks as it relates to the amount of detail within the Unit 1 FSAR. As discussed and accepted by the NRC, once the cask loading pit gate is set in place, the spent fuel pool is isolated and the cask loading pit is no longer considered part of the SFP (0CNA070301). The FSAR describes the methodologies and the various accident conditions that are related to the Unit 1 Region 1 and Region 2 racks. The HI-STORM 100 cask differs in some of the discussions related to the Region 1 and Region 2 racks as such a full Evaluation shall be performed. Additionally, a new subsection is being added in ANO-1 SAR 9.6 to only discuss the criticality analysis for the cask loading pit while the pit gate is open. Details of the Region 1 and Region 2 racks are also described within the TS bases and as such the loading of a HI-STORM 100 cask will impact these discussion as well which will warrant a 50.59 evaluation to be performed.

The analysis considered all the normal and accident cases discussed in the Unit 1 FSAR and the Unit 1 Bases as they related to SFP criticality. The analysis was performed in accordance with a poison rack analysis which is similar to the Region 1 SFP racks. The Unit 1 FSAR describes the criticality codes used to analyze the cask. The same codes used to analyze the Region 1 racks were used to evaluate the HISTORM 24 assembly cask with the exception that an additional code was used. The primary code for the new cask criticality calculations was MCNP, a Monte Carlo code developed by the Los Alamos National Laboratory. KENO-V, the code originally used, was also used but because it tended to produce less conservative results the MCNP runs were used.

As mentioned in the review of TS 4.3.1.1 b., the Unit 1 FSAR lists the uncertainties and tolerances considered in the SFP Region 1 criticality analysis. The FSAR states that the tolerances can be treated as worst case or by statistically combining the reactivity associated with each tolerance. All the tolerances listed in the Unit 1 FSAR for the racks were considered and their positive reactivity effects were evaluated. Some of the fuel assembly tolerances used in the analysis used more reactive parameter than originally evaluated.

The original criticality calculation for the Region 1 racks used Boraflex as the neutron absorber material. Credit was taken for the neutron absorption of all fuel length neutron absorption materials which included the rack structural materials and any material used for the purpose of absorbing neutrons. The difference being is the HI-STORM 100 cask utilizes Boral or Metamic as a neutron absorber. These materials have both been approved for use in spent fuel pools by the NRC. All the failure modes of the newer materials are the same where gamma and chemical attack will degrade these materials over the long term storage within the spent fuel pool. These failure modes from chemical attack and radiation exposure have been improved upon with the use of aluminum as a matrix material over silicon rubber. As such the poison panels did not consider end shrinkage or gap formation because these are not failures identified to occur with the newer materials. Additionally 100% of the $_{10}B$ content was assumed to be present in the analysis. In keeping with the Unit 1 FSAR poison material evaluation requirements the minimum dimensions and the minimum $_{10}B$ content was used in the analysis.

No neutron leakage was assumed to occur as originally assumed in the Unit 1 Region 1 rack analysis.

The Unit 1 FSAR states that the water was treated as 68°F. A more conservative approach was used where it was assumed that under normal storage condition the SFP temperature would be at the most reactive condition at 273K. This positive reactivity effect was treated as an additional bias.

The Unit 1 FSAR states the no credit was taken for fuel burnup in Region 1. The HI-STORM 100 24 assembly cask was evaluated for unrestricted storage of fresh fuel assemblies with no credit for fuel burnup. Additionally, in keeping with the original assumption in the Unit 1 FSAR $_{234}U$ and $_{236}U$ were not included in the fuel composition.

Minor structural members that did not extend the fuel length of the active fuel region were not included in the model as stated in the Unit 1 FSAR. This included fuel assembly end fittings, the fuel grids, and the cask bottom support plate.

The fuel assembly analyzed was a B&W 15x15 fuel assembly where no radial enrichment zoning was credited. The fuel assembly enrichment analyses were at 4.10 wt% or greater. No credit was taken for the storage of control components stored within the fuel assemblies.

No credit was taken for soluble boron present in the water for the normal storage conditions.

The results of the HI-STORM 100 24 assembly cask analysis showed that with a 95 percent probability and a 95 percent confidence level the k_{eff} was less than 0.95 as required by the Region 1 SFP rack analysis.

The Unit 1 FSAR and TS bases discuss a number of accident conditions that could affect reactivity within the spent fuel pool. When performing the analysis for postulated accidents in the SFP, the double contingency principle of ANS N16.1-1975 was applied. This states that it is unnecessary to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Therefore, for accident conditions, the presence of soluble boron in the SFP water can be assumed as a realistic initial condition since its absence would be a second unlikely event.

The fuel assembly accident cases evaluated within the FSAR are the following, fuel drop horizontally on a cask, fuel drop on a fuel assembly, fuel drop next to a cask, a fuel drop on the cask basket, and fuel misplacement. Additionally loss of SFP cooling and seismic events were considered. The only event that was shown to effect criticality was the drop of a fuel assembly on the basket. The analysis was performed in accordance with the methodology used to evaluate this event for the Region 1 SFP racks. The analysis assumed that all the poison material was damaged and replace with water. The table below illustrates the various accident cases that effect reactivity within the ANO-1 SFP. The matrix provides analytical results for the boron concentration that is required to ensure k_{eff} remains below 0.95 for the specific accident. As required, in all cases the minimum boron concentration (1600 ppm) required by ANO-1 TS 3.7.14 bounds the analytically determined soluble boron concentration required to protect against the postulated accident.

Table 1

Criticality Analysis	TS Boron Concentration (ppm)	Criticality Assumptions $k_{eff} \leq 0.95$	Misloading Accident Analysis Boron Concentration (ppm) $k_{eff} \leq 0.95$	Dropped Fuel Assembly on the ousted of the Rack/Cask	Misalignment of Active Fuel Region
SFP Region 1 Part 50	≥ 1600	No credit for boron	0	0	1600
SFP Region 1 Part 50 (MPC-24 not included)	≥ 1600	No credit for boron	1600	1600	1600
MPC-24 Part 50	≥ 1600	No credit for boron	0	0	800
MPC-24 Part 72	0	No credit for boron	n/a	n/a	n/a

TRM 3.7.8 addresses SFP boron sampling requirements. This section will be modified to include the cask loading pit in the applicability section. The COLR and the Unit 1 SERs related to FSAR changes are not impacted by changes related to the SFP boron concentrations or the storage of fuel in a HI-STORM 100 cask system. The Unit 1 SERs for amendments that discuss in detail the stored fuel in the Unit 1 SFP do not impact the use of the HI-STORM 100 24 assembly cask in the Unit 1 SFP cask pit.

LBDs Controlled Under Other Regulations:

The QAPM, emergency plan, fire protection program, and the ODCM discussion of the storage of fuel are not impacted by the use of the HI-STORM 100 24 assembly cask system related to the site specific Unit 1 criticality calculations. No changes to these LBDs are needed.

Test and Experiment:

The HI-STORM 100 criticality analysis does not cause components or structures to be used outside its design bases and does not require operation of equipment outside the conditions analyzed in the FSAR. It is therefore not a test or experiment as defined by LI-101 and can not be a test or experiment not described in the FSAR.

ISFSI

Criticality analysis when fuel is loaded in the cask is already in compliance with Part 72. This analysis is related to the HI-STORM 100 system. However, a 72.48 Screening will be performed in accordance with LI-112 since the first checkbox in the IFSFI Screening is applicable.

4. **References**

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.5.1[5](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
Autonomy – 50.59 Common	ALL(B&W 15x15, boron concentration, fuel burnup, burnup, dry fuel storage, ISFSI, MPC*, cask, shipping cask, fuel w/20 storage, region w/20 pool, pool w/20 enrichment, cask and “enrichment”, pool w/30 cooling time, pool and “cooling time”, cask w/30 licensing, “dry fuel storage”; fuel w/20 24; storage w/20 burnup, ‘boron credit”, spent fuel pool w/20 criticality, spent fuel pool w/20 boron, “50.68”, “70.24”, physics parameters, fuel parameters, fuel design, boral, metamic, boraflex

LBDs/Documents reviewed manually:

Unit 1 TS 3.7.14, 3.7.15, 4.3.1

Unit 1 TS Basis 3.7.14 and 3.7.15

Unit 1 TRM 3.7.8

Unit 1 FSAR 3A.5, 9.6, 9.10, 11.3

0CNA109805, 0CNA070301, 0CNA048314, 0CNA060304

5. **Is the validity of this Review dependent on any other change?**

☐ **Yes**
☒ **No**

If “YES”, list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure ENS-EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure ENS-EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING
(NOTE: This section is not applicable to Waterford 3 and may be removed from 50.59 Reviews performed for Waterford 3 proposed activities.)

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure ENS-LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	Yes	No	
1.	☒	☐	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: _____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions.

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☒ Yes ☐ No

The only change being performed under this 50.59 is a change to the ANO-1 criticality analysis for the Holtec HI-STORM 100 -24 cask design. Only Question 8 is necessary to be answered, however, the other 7 questions are being discussed for completeness.

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The criticality analysis of the 24 assembly cask HI-STORM 100 is being analyzed in the same manner as a spent fuel pool fuel storage location. The evaluation of the cask similarly to the spent fuel pool criticality analysis does not result in the use of fuel handling equipment in any different configuration or interface than the fuel handling equipment was originally designed for, such as loading spent fuel shipping cask in the spent fuel pool cask pit or anywhere else in the pool.

   Fuel Handling Accident

   The fuel assemblies are stored under water in the spent fuel storage pool and are moved to the cask after independent verification of pool and cask pit boron concentration. Both the storage racks and cask have a safe geometric spacing and or fixed neutron poison. Under these conditions, a criticality accident during refueling or cask loading is not considered credible. Mechanical damage to the fuel assemblies during transfer operations is possible. A mechanical damage type of accident is considered the maximum potential source of activity release during refueling or cask loading operations.

   The assumptions made for this analysis are shown in Table 14-24 of the FSAR. The reactor is assumed to have been shut down for 100 hours, since Part 50 Technical Specifications prohibit fuel handling operations prior to this time. Fuel assemblies have to be moved to the spent fuel pool to allow fission products to decay sufficiently prior to being loaded in a storage cask. Casks loading operational requirements specify a minimum fuel assembly cooling time of 3 years or more. It is further assumed that the cladding of six rows of fuel rods in the assembly, 82 of 208, suffers mechanical damage. The cask configuration is sufficiently similar to spent fuel racks as to not induce additional assembly damage; therefore, the fuel handling accident is bounded as described in FSAR Section 14.2.2.3 for cask loading. This includes fuel handling accident for four drop scenarios (fuel drop horizontally on a cask, fuel drop on a fuel assembly, fuel drop next to a cask, and a fuel drop on the cask basket) and these are unaffected by the criticality calculations performed for cask loading activities. The same equipment and procedural controls for controlling fuel within the spent fuel pool are utilized when loading fuel in the cask.

   The temperature effects on the fuel assemblies in the cask were evaluated and the fuel assemblies in the cask were evaluated for the most reactive temperature for all accident and normal storage cases. The loss of SFP cooling will have no impact on the cask criticality analysis because the temperature coefficient of reactivity is negative and the introduction of voids will further decrease reactivity.

   The cask is analyzed for loading all fresh fuel up to the Unit 1 spent fuel pool enrichment limit of 4.1 wt% and it was shown that the cask satisfies the requirement to have a k-effective that does not exceed 0.95, at a 95% probability, with a 95% confidence level. Also, loading fuel in the cask will have no impact to the boron dilution event probability. The same controls for prohibiting a dilution event during spent fuel movement activities are in use when loading/unloading a cask within the cask pit. As such boron was credited for mitigating the reactivity effect for a dropped assembly where the boron requirement was determined to be 800 ppm which is below the TS limit of 1600 ppm. Therefore, as the parameters that would increase the probability of accidental criticality are not changed from that previously analyzed, loading of a cask in the cask loading pit does not increase the frequency of occurrence of criticality events in the SFP or cask loading pit evaluated in the FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The calculation only changes the way the HI-STORM 100 cask is treated from a criticality perspective. The cask analysis will not affect failure mechanisms of the fuel handling equipment or the SFP (walls, floors, racks, cooling, etc.). There is no difference in how the spent fuel equipment or pool structure is utilized for pool or cask loading pit fuel movement operations. The fuel handling equipment and the SFP will perform as designed and described in the FSAR. The fuel handling equipment is designed to retrieve a fuel assembly from one location and place it in another. The re-located positions can be other locations in the spent fuel racks, in the upender for transfer to the reactor building or in the cask loading pit. However, for cask fuel loading in the cask loading pit, the gate is placed between the pool and the cask loading pit once the cask has been loaded to capacity. This could allow un-likely boron dilution in the pit at an accelerated rate from a hypothetical uncontrolled source of unborated water in the spent pool area such that the minimum boron concentration in the pit water is not maintained. However, subcriticality is maintained because the fuel assemblies are independently verified to be correct and properly located in the cask prior to gate closure, and like evaluated in the pool rack criticality analysis, the cask analysis evaluated the fuel without crediting soluble boron. In addition, as evaluated for the SFP, heat up of the isolated cask loading pit from 24 maximum burned fuel assemblies is sufficiently slow such that cask loading pit thermal limits are not approached during the cask loading evolution. Should extended isolated cask loading pit operation be required, the gate can be removed or cooler pool water pumped into the pit (the gate allows leakage back to the pool such that the levels are maintained near level). These scenarios are not different from those necessary for use of the originally described use of a shipping cask in the cask loading pit.

Therefore, this change does not increase the likelihood of the spent fuel handling equipment or structure malfunction.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The criticality requirements of the SFP and the cask loading pit ensure that there is adequate boron concentration in the pool to prevent an accident. The performance of a criticality analysis for the fully loaded cask does not perform an accident mitigation function. The fuel assembly design, enrichment limits, weight, and tolerances were the same as in the original analysis. The consequence of damaging a fuel assembly and the racks remains the same due to a fuel assembly drop. The worst case fuel misplacement in the cask showed that the boron requirements are less than the original requirement of 1600 ppm to keep reactivity below 0.95 in the pool racks. None of the accidents postulated in the FSAR from criticality, fuel assembly damage, or pool structural damage are increased as a result of putting fuel in the cask instead of a fuel pool rack. The mechanisms for potential damage are sufficiently similar or exactly the same such that increased consequences are not possible or predicted

Therefore, this change does not increase the consequences of any accident.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The fuel handling accidents evaluated in the FSAR for the pool racks extended to the cask are assembly drop on the cask, onto another fuel assembly, next to the cask, onto cask cell, fuel misplacement, and boron dilution. The fuel assembly design, enrichment limits, weight, and tolerances were the same or bounding as in the original analysis for the spent fuel pool racks as discussed in the FSAR. However, the cask cell assembly interface with the fuel assembly is different. There are bars that extend above the cask cells for structural support of the cask lid during a loaded cask transportation accident. If the fuel assembly were dropped onto a bar, it is expected that fuel rods damaged will not exceed that previously evaluated for the drop of an assembly due to the size of the bars and fewer fuel rods would be impacted (the analysis does not credit protection from the end fittings). Also, the configuration of the edge of the cask is sufficiently similar to the edge of a rack such that increased damaged is not predicted. Therefore, the consequences resulting from a damaged fuel assembly on the racks, the cask, or the pool structure remain the same.

Dropping a fuel assembly causing a misplacement accident was reanalyzed for the HI-STORM 24 assembly cask taking into consideration the specific geometry changes associated with the cask. The worst case fuel misplacement showed that the boron requirements for the cask racks are less than the original pool rack analysis requirement of 1600 ppm to keep reactivity below 0.95.

Therefore, this change does not increase the consequences of any malfunction of a structure, system or component important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The analysis of the HI-STORM 100 cask considered all the accidents postulated for the SFP racks (fuel drop on the cask, fuel drop on fuel assembly, fuel drop next to the cask, fuel drop on the cell, fuel misplacement, and boron dilution). The cask has the same basic physical configuration of a rack where fuel assemblies are stored in vertical cells in a grid with a flux trap separation between each cell and the poison material is located on the outside of each cell. The cask cell walls are thicker than the racks, the outside wall is thicker than the racks and the space for mishandling is tighter than around the racks. The cask loading and configuration does not introduce any new rack related events that have not been previously analyzed when the cask pit is opened to the SFP. After the gate is closed and the pit isolated from the pool, heat up of the isolated cask loading pit from 24 maximum burned fuel assemblies is sufficiently slow such that cask loading pit thermal limits are not approached during the cask loading evolution and is not different from that necessary for use of the originally FSAR described use of a shipping cask loading in the cask loading pit. Once the cask loading pit gate is closed, the cask and associated analyses no longer have to be considered under Part 50, but only under Part 72. The use of a cask in the cask pit has no physical effects on the racks or abnormal effect on the fuel handling equipment. The rack accidents as discussed in the FSAR and TS bases as listed above were analyzed for loading activities in the cask. The analyses demonstrated that the rack accidents described in the FSAR and TS Bases bounded the results of the cask handling analyses.

Therefore, this change does not create the possibility for an accident evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

All the cask loading criticality analyses results from the SFP rack accidents (fuel drop on the cask, fuel drop on fuel assembly, fuel drop next to the cask, fuel drop on the cell, fuel misplacement, and boron dilution) are bounded by the same accidents as postulated in the FSAR and TS bases. The cask loaded with fuel sitting in the cask loading pit during a seismic event could result in cask impact on the cask loading pit wall, but does not have different results for the pool structure or fuel assembly integrity than that was previously evaluated for the pool racks

Therefore, this change does not create the possibility of causing a component important to safety from malfunctioning.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

The fuel assembly design, enrichment limits, weight, and tolerances were the same or bounding as in the original analysis. Fuel cladding which is a fuel fission product barrier is unaffected by the analysis of the 24 assembly HI-STORM 100 cask as a SFP rack. In addition, all fuel movement continues to be under water until the cask lid is welded which provides an additional fission product barrier.

Therefore, this change does not affect any fission product barriers.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☒ No

BASIS:

The RIS infers that the Part 50 criticality requirements should be imposed on the dry fuel storage cask when the cask is in the SFP. It is stated that the cask should be treated/analyzed as a spent fuel pool rack from the perspective of imposing the spent fuel pool criticality requirements for normal storage and the requirements for addressing the spent fuel pool accident cases. Since the Unit 1 24 assembly HI-STORM 100 cask criticality calculations are not explicitly detailed in the Unit 1 FSAR, comparisons can be made with the detailed discussions on how the criticality calculations are preformed on the spent fuel pool racks for Region 1 and Region 2. The justification for the four changes to the way the original analysis was performed is provided below.

Absorption Material - The neutron absorbing material is different from that currently contained in the ANO-1 spent fuel pool; however, the absorption characteristics are similar to that of Boraflex in the fact that boron is suspended in a matrix material. The purpose of this 50.59 Evaluation is not to license Boral/Metamic for the spent fuel pool, but only to ensure that criticality analysis being performed meets the current ANO-1 licensing basis requirements under Part 50. The use of a poison material to off set reactivity is the basic method addressed in the Unit 1 FSAR and the use of these other NRC approved neutron absorbers in a HI-STORM 100 cask does not differ from that methodology for controlling reactivity.

Analysis Design Parameters – The change in parameters included such things as greater fuel density, larger fuel pellet OD, flooded pellet cladding water gap, bounding enrichments used in normal and accident cases, and evaluated positive reactivity effects for temperatures below 68°F under normal storage conditions. Even though some of the design parameters have changed from that of the spent fuel pool analysis, none of these parameters represent a departure from a method approved by the NRC. Per the guidance in Section 4.3.8 of NEI 96-07 (accepted in Regulatory Guide 1.187), "*Use of a methodology revision that is documented as providing results that are essentially the same as, or more conservative than, either the previous revision of the same methodology or another methodology previously accepted by NRC through issuance of an SER"* are not considered departures from a method of evaluation. In addition in section 3.8 of NEI 96-07 an Input Parameter is not a change in methodology as the guidance states: *if a licensee opts to use a value more conservative than that required by the selection method, reduction in that conservatism should be evaluated as an input parameter change*. The changes to the criticality analysis use the same methodology as that contained in the spent fuel pool analysis. None of the input parameters were specifically accepted by the NRC staff as parameters that could not be changed under 10 CFR 50.59.

Code methodology - Even though the ANO-1 SAR references the KENO computer code, MCNP is a code widely accepted by the NRC to be used in spent fuel pool criticality applications and the industry has utilized the code extensively. The code has extensive benchmarking comparisons against test data, plant data, and other approved codes such as KENO-V. Section 3.4 of NEI 96-07 defines a departure of a methodology. Included in that definition is the ability to change from one method to another method without being a "departure." *A licensee may adopt completely new methodology without prior NRC approval provided the new method is approved by the NRC for the intended application. A new method is "approved by the NRC for the intended application" if it is approved for the type of analysis being conducted and the licensee satisfies applicable terms and conditions for its use.* Therefore, there is no departure from a methodology by changing from the KENO code to the MCNP code.

Therefore, this change does not depart from the evaluation methods outlined in the FSAR.

If any of the above questions is checked "YES", obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure ENS-LI-113.

ANO 50.59 Evaluation Number

2005-025

I. OVERVIEW / SIGNATURES

Facility: ANO - Unit 1

Document Reviewed: ER-ANO-2002-1078-016, ANO-1 SG/RVCH Replacement - Insulation
Change/Rev.: 0

System Designator(s)/Description:

Description of Proposed Change:

In support of the Arkansas Nuclear One, Unit 1 Steam Generator / Reactor Vessel Closure Head (SG/RVCH) Replacement, ER-ANO-2002-1078-016 "ANO-1 SG/RVCH Replacement – Insulation", provides for the permanent removal of existing insulation from the Once Through Steam Generators (OTSGs) E-24A and E-24B and subsequent installation of replacement metal reflective insulation. ER-ANO-2002-1078-016 also addresses the installation of metal reflective insulation for various piping systems within the scope of the ANO-1 SG/RVCH Replacement Project.

Due to the configuration and dimensional variances between the Original Once Through Steam Generator (OOTSG) and the Replacement Once Through Steam Generator (ROTSG) (i.e. new inspection ports, instrument nozzle repositioning, thermocouple mounting pad repositioning and alteration), the insulation on the OOTSG is to be discarded.

A new insulation system consisting of metal reflective panels and the necessary means for its support will be installed on the ROTSG including the nozzles. The new insulation will incorporate changes in the ROTSG configuration.

ER-ANO-2002-1078-016 also provides for replacement of insulation on other piping systems within the scope of the ANO-1 SG/RVCH Replacement Project where existing insulation to be removed is non-metallic or where existing metal reflective insulation is damaged to the extent that it is required to be replaced with new metal reflective insulation (MRI). Conventional insulation on Main Steam and Main Feedwater lines and the Emergency Feedwater nozzles will be removed by ER-ANO-2002-1078-012 "ANO-1 SG/RVCH Replacement – Main Steam Piping & Supports" and ER-ANO-2002-1078-013 "ANO-1 SG/RVCH Replacement – Main Feedwater & Emergency Feedwater Piping & Supports". Fibrous insulation on portions of the RCS Hot and Cold leg elbows will be removed by ER-ANO-2002-1078-015 "ANO-1 SG/RVCH Replacement – OTSG Removal & ROTSG Preparation/Installation". The new MRI will be installed by ER-ANO-2002-1078-016. The connected piping systems include:

- Main Steam piping from the Main Steam nozzle to the SG cavity wall penetration
- Main Feedwater piping from the Main Feedwater header to the SG cavity wall penetration
- Emergency Feedwater (EFW) nozzles and a short piece of connected EFW piping connected to the OOTSG EFW nozzles
- RCS Hot Leg 180° elbow
- RCS Cold Leg

The existing fibrous insulation on the steam generator bottom head, Main Feedwater header (feedring and risers), and Emergency Feedwater header nozzles along with the MRI on the remainder of the OOTSG and its associated support system will be removed and appropriately disposed.

The ROTSGs will be insulated with MRI following installation in each of their respective cavities. Some preliminary work such as installation of the support rings and installation of the bottom head insulation may be performed with the ROTSGs in the staging area outside of the Reactor Building. The MRI is designed to provide access for in-service inspection of welds and for general equipment inspection and maintenance activities. All panels are fully removable for inspection and maintenance purposes.

The thermal insulation on piping and components in the Reactor Building can potentially be dislodged by a high energy line break, creating loose debris that could potentially block flow paths in the Reactor Building including the reactor building sump. Because of its higher density, MRI tends to settle and not migrate to flow blockage areas. ER-ANO-2002-1078-016 will reduce the quantity of non-metallic thermal insulation currently in the Reactor Building.

The MRI will be supported by an insulation support system on the ROTSG. The structural models used in the loading and stress analysis of the RCS in ANO Calculation 021381E101-62 “ANO-1 SGR RCS Structural Model” (AREVA Calculation 32-5017972-02, included in ER-ANO-2002-1381-000 “ANO-1 ROTSG Design/Qualification”) includes the weight of the insulation. Although the weight associated with the replacement insulation system is significantly higher (approximately double (ANO Calculation ANO-ER-05-002)) than the weight of the existing insulation system (ANO Calculation 92-D-5005-05), the structural models used in the loading and stress analyses of the RCS include the weight of the replacement insulation system and remain bounded by the combined ROTSG and insulation weight assumed in ANO Calculation 021381E101-62. The structural evaluation of the ROTSG provided acceptable results, as documented in ANO Calculation 021381E101-68 “ANO-1 SGR RCS Structural Evaluation” (AREVA Calculation 32-5017980, included in ERANO-2002-1381-000).

A seismic evaluation of the MRI installed on the ROTSGs was performed by the insulation vendor (Transco) and found acceptable (ANO Calculation ANO-ER-05-002, “Structural Analysis of Metal Reflective Insulation Support Systems for the Steam Generator Replacement Project at Arkansas Nuclear One, Unit 1” (Transco Calculation RG-49112-TCR2)). This seismic evaluation determined that the insulation would remain in place following a seismic event.

The replacement MRI for ANO-1 ROTSGs and piping is designed to limit heat loss to an average rate of 55 BTU/hr·ft^2 for the entire insulation system. The thermal performance of the new MRI is evaluated in ANO Calculation ANO-ER-05-001 and is within the specified maximum of 55 Btu/hr-ft^2. From ANO Calculation M-3600-01 the insulation heat loss from the OOTSGs is estimated as 188,000 BTU/Hr. From ANO Calculation ANO-ER-05-001, the total heat loss from the newly insulated ROTSGs is less than 188,000 BTU/Hr. Therefore, the calculated heat load inside the Reactor Building is reduced by replacing the steam generator insulation.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-025)**	**Sections I, II, and IV required**

Preparer: Doug Barborek / **ORIGINAL SIGNED BY DOUG BARBOREK** / EOI / SG-RVCH / 7-21-05
Name (print) / Signature / Company / Department / Date

Reviewer: John Pearman / **ORIGINAL SIGNED BY JOHN PEARMAN** / EOI / SG-RVCH / 7-21-05
Name (print) / Signature / Company / Department / Date

OSRC: J.R. Eichenberger / **ORIGINAL SIGNED BY J.R. EICHENBERGER** / 7-21-05
Chairman’s Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	Table 4-12
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☐ **Yes**
☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. Basis

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

During the SG/RVCH Replacement outage, the existing calcium silicate, fibrous, and metal reflective insulation (MRI) on the OTSGs and attached piping as described in Section I of this 50.59 Review will be replaced with new MRI. The new MRI is designed to maintain or improve the current insulation design basis including RCS heat loss, free draining, accessibility of components for inspection, and reduction of sump clogging materials.

Operating License/Technical Specifications (TSs)

The ANO-1 Operating License, Technical Specifications, Technical Specification Bases, and Technical Requirements Manual were reviewed to determine the impact of the proposed replacement of the insulation on the OTSG and connected piping. Insulation is not discussed in the Operating License, Technical Specifications, Technical Requirements Manual, or Core Operating Limits Report. As discussed in Tech Spec Bases 3.4.9, the pressurizer heaters are used to maintain pressure in the RCS so that reactor coolant in the loops is maintained subcooled. This function must be maintained during a postulated loss of offsite power. The pressurizer heater capacity is based on the expected heat loss from the RCS. With the increased thickness of the new MRI and expected better panel fit, the RCS heat loss through the ROTSGs to the Reactor Building atmosphere will be less than the current heat loss.

Therefore, no changes to the ANO-1 Operating License, Technical Specifications, Technical Specification Bases, or Technical Requirements Manual are required.

FSAR

The RCS insulation is discussed in a number of sections in the FSAR. FSAR Section 4.2.2.7 notes that some fiberglass insulation has been installed on the RCS. Although ER-ANO-2002-1078-016 replaces some of the fiberglass insulation with MRI, fiberglass insulation currently installed in other locations on the RCS is not impacted by this ER. Section 4.2.2.7 further notes that the existing insulation provides free drainage of moisture and condensate. These criteria are applicable to the new MRI as well. Therefore, FSAR Section 4.2.2.7 does not need to be revised.

FSAR Section 4.4.1 notes that the steam generator and RCS piping external surface are accessible for inspection. The new MRI panels are individually removable; therefore, no changes are needed to this section.

FSAR Table 4-12 lists the OTSG weldments. Item 2.8.14 lists the Insulation Support Lug Pads. The new MRI will be attached to each steam generator by friction support rings that do not require welded attachment to the steam generator shell. Therefore, Item 2.8.14 should be deleted.

Fire Hazards Analysis

The Fire Hazards Analysis for the Unit 1 Reactor Building in the area of the OTSGs identified combustibles in the area to include lube oil, flame resistant cable insulation, jacketing material, and transients. ER-ANO-2002-1078-016 will replace non-metallic insulation on the OTSGs and connected piping, replacing it with new MRI. The MRI is not combustible, will not create any new ignition sources, and does not absorb fluids such as oil. The measures intended to minimize the probability of a fire have not changed, and the probability of a fire will not be increased as a result of this ER Response. The Fire Hazards Analysis does not need to be revised.

Test or Experiments Consideration

The proposed modification will replace the insulation on the steam generators and connected piping. It will maintain the current design basis function of minimizing heat loss to the Reactor Building atmosphere, providing free drainage of moisture and condensate, and not creating sump blockage issues. Replacement of the existing insulation will not result in the Reactor Coolant System being used for functions that are outside of its design basis. Replacement of the insulation on the steam generators and connected piping is not considered a test or experiment.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LRS 50.59 – Unit 1	insulation w/20 fire, insulation w/10 fib*, insulation w/10 steam generator, insulation w/10 OTSG, insulation w/10 metal, insulation w/10 sump, insulation w/10 oil, insulation w/10 loss*, insulation w/10 heat load, insulation and net free volume, insulation and heat sink, insulation and npsh, calcium, cal-sil, MRI, sump w/10 blockage, heat loss*, RCS w/10 heat load, reactor w/10 heat load, NSS* w/10 heat load, 1.82, steam generator w/10 heat lo*, insulation w/20 helb, insulation w/20 melb, insulation w/20 energy, insulation w/20 break, insulation w/20 missile

LBDs/Documents reviewed manually:	
ANO Unit 1 FSAR	Sections 4.2.2.7, 4.4.1, 6.2, 9.8, Apps 9A and 9D, and A.7.3.17; Chapter 14; Tables 4-4 and 4-12
ANO Unit 1 Technical Specifications, Technical Specification Bases, and Technical Requirements Manual	T.S. Section B 3.4.9
ANO Units 1 and 2 – Fire Hazards Analysis	FHA Sections 1 thru 8, Section 9 (for Area J)

5. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☐ **Yes**
☒ **No**

If "YES," list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: _____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10 CFR 50.59 Review Program Guidelines for guidance.

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.	☐ Yes ☒ No

Does the proposed Change:

1.	Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?	☐ Yes ☒ No

BASIS:

ER-ANO-2002-1078-016 provides for the removal of metallic and fiberous insulation from the OOTSGs in support of steam generator replacement. Insulation will be removed from connected piping systems (hot and cold leg RCS piping, Main Steam, and Main Feedwater) by their respective ER-ANO-2002-1078-series ER Responses. New metal reflective insulation (MRI) will be procured and installed on the ROTSGs, hot and cold leg RCS piping, and other connected piping by implementation of ER-ANO-2002-1078-016. Thermal Insulation installed on the affected systems is not considered an accident initiator. ER-ANO-2002-1078-016 will be performed with Unit 1 in Modes 5, 6, or defueled. Some preparatory work such as installation of the support rings may be performed before the ROTSG is moved to the Reactor Building. The only FSAR Chapter 14 event that occurs with the Unit shut down is a fuel handling accident in the Reactor Building or in the spent fuel pool. The insulation removal and replacement activities performed in ER-ANO-2002-1078-016 will not increase the probability of a fuel handling accident.

Of concern is the insulation on the ROTSG and connected piping potentially being displaced in a seismic event and breaking instrument lines or small bore piping and in so doing creating a small RCS pressure boundary leak. The replacement MRI will be similar to the existing MRI with the exception that the replacement MRI panels for the ROTSG will be thicker, thereby increasing the weight of the insulation on the ROTSG. The new MRI panels will be supported by friction support rings and bolted attachments to the ROTSG. This arrangement allows for the thermal growth of the ROTSG. A seismic evaluation (ANO Calculation ANO-ER-05-002, "Structural Analysis of Metal Reflective Insulation Support Systems for the Steam Generator Replacement Project at Arkansas Nuclear One, Unit 1" (Transco Calculation RG-49112-TCR2)) was performed using the increased weight of insulation and attachment method and found to be acceptable. Therefore, the replacement insulation would remain in place following a seismic event and not come loose and potentially damage instrument lines and small bore piping.

Although the weight associated with the replacement insulation system is greater than the weight of the existing insulation system, the structural models used in the loading and stress analyses of the RCS include the weight of the replacement insulation system and remain bounded by the combined ROTSG and insulation weight assumed in ANO Calculation 021381E101-62.

The new piping MRI replaces existing calcium silicate or fiberous insulation on connected piping under the scope of ER-ANO-2002-1078-016. The stress analyses for these piping systems have been revised to include the change in weight due to the new MRI. The stresses in these piping systems remain within the ASME Section III Code.

Therefore, it is concluded that the changes in ER-ANO-2002-1078-016 would not increase the frequency of an accident previously evaluated in the FSAR, neither during implementation nor during subsequent plant operation.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

ER-ANO-2002-1078-016 will install MRI on the ROTSG, a portion of the RCS hot and cold leg piping, and on Main Steam and Feedwater piping and the Emergency Feedwater nozzles. The new MRI, like the current MRI, is fabricated from stainless steel. The new MRI will therefore have similar physical properties. It is not combustible, it is not chemically reactive, and it is not readily transportable.

The thickness of the MRI to be installed on the ROTSGs will be increased from the current four inches (nominal) to five inches (nominal). This increase in thickness was necessary to achieve the design basis heat loss of 55 BTU/hr-ft^2. The increase in thickness of the insulation panels results in an increased weight of insulation on the ROTSG. As discussed in response to Question 1, this increased weight was accounted for in the seismic analysis performed by Transco (ANO Calculation ANO-ER-05-002) and was used in the loading and stress analyses of the RCS performed by AREVA (ANO Calculation 021381E101-62). All of these analyses produced acceptable results. The increased thickness of insulation on the ROTSGs and connected piping was evaluated relative to potential impacts on the net free volume and heat sink assumptions used in the Reactor Building pressure and hydrogen generation analyses. These analyses are described in ANO Engineering Report 94-R-1018-01, ANO-1 Reactor Building Volume and Internal Surface Area Criteria. As noted in this report, heat transfer to insulated surfaces is ignored and as a result, the new MRI will not impact the heat sink assumptions utilized for Reactor Building pressure analysis. The increased thickness of the the ROTSG and piping MRI will have a negligable impact on the net free volume, since the MRI is vented and its crushed thickness is less than the solid thickness of the insulation currently assumed in ANO Engineering Report 94-R-1018-01.

Installation of new MRI on the ROTSGs and connected piping does not change the operation or function of the thermal insulation or increase the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The new ROTSG and connected piping insulation was evaluated relative to impacts on post-LOCA doses. Reactor Building post-LOCA leakage is directly related to Reactor Building pressure. The Reactor Building net free volume and heat sinks are assumptions in the FSAR Chapter 14 LOCA peak pressure analysis (FSAR Figure 14-61). As discussed in response to Question 2, the increased thickness of the MRI will have a negligible impact on the net free volume and no impact on the heat sink assumption. The increase in thickness of the MRI would not adversely impact the volume and the Reactor Building pressure analysis, and therefore no change in the offsite dose would be expected.

No new source term is added and no equipment assumed to mitigate dose events is affected. The planned installation of new MRI on the ROTSGs and connected piping has no adverse effect upon the accident mitigation capability of any SSC and therefore will not alter the consequences (increase radiological dose) of any accidents previously described in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

ER-ANO-2002-1078-016 will install new MRI on the ROTSGs and connected piping, serving to replace existing MRI, calcium silicate and fiberous insulation. Calcium silicate and fiberous insulation represent a greater risk for sump blockage post-LOCA than MRI. MRI is less readily transported to the Reactor Building Emergency sump. The effect of the loss of the sump for cooling is not evaluated in the FSAR for dose consequences. The replacement of a portion of the calcium silicate and fibrous insulation currently in the Reactor Building with new MRI is considered a safety enhancement. As discussed in response to Question 2, no SSCs important to safety are affected. Therefore, any consequences as a result of a possible malfunction of any structure, system, or component (SSC) important to safety previously evaluated in the FSAR, will not be increased.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The replacement MRI meets all design requirements for insulation based on applicable Code sections and FSAR description. The replacement of the existing non-metallic and MRI with new MRI will not affect the operation of the plant. As discussed in response to Question 1, the weight and attachment method for the new MRI to be installed on the ROTSGs is different from the current MRI on the OOTSGs. A seismic evaluation of the new MRI configuration was performed and found to be acceptable. The new MRI conservatively supports or maintains current RCS heat loss assumptions for the plant.

Therefore, this activity will have no impact on the existing accident analyses in the ANO-1 FSAR and does not create a different type of accident.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The replacement of the existing non-metallic and MRI with new MRI will have no effect on equipment important to safety and will in no way affect the way the plant responds to accidents. There is no Failure Modes and Effects Analysis for thermal insulation discussed in the FSAR. The replacement MRI meets all requirements for thermal performance, seismic support, fire loading, chemical reactivity, and radiation aging as required by Specification ANO-M-136. Therefore, this ER Response would not have created the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

ANO-1 fission product barriers include fuel cladding, RCS Boundary, and Containment Pressure.

Fuel Cladding

As discussed in FSAR Section 14.2.2.5, Loss of Coolant Accident, long term core cooling assumes that the LPI pump suction is switched from the BWST to the Reactor Building sump. If the sump became clogged with debris from component or piping insulation, the sump performance could be compromised. Replacement of the calcium silicate and fibrous insulation currently on the OOTSG lower head and connected piping with MRI reduces the potential for sump blockage since MRI is less likely to migrate to the emergency sump due to its greater weight. Therefore, the proposed removal and replacement of thermal insulation will not result in a design basis limit for the fuel cladding barrier as described in the FSAR being exceeded or altered.

RCS Boundary

The new insulation on the ROTSGs and connected piping was evaluated relative to being displaced in a seismic event and potentially breaking instrument lines or small bore piping thereby creating a small RCS pressure boundary leak. As discussed in response to Question 1, a seismic evaluation was performed that demonstrated that the new MRI would not come loose in a seismic event. Therefore the RCS pressure boundary will not be adversely affected by the proposed ER.

Containment

As discussed in the response to Question 2, the installation of new MRI will not adversely impact the Reactor Building pressure analyses. Specifically, the installation of new MRI will not impact the heat sink assumptions and will have a negligible impact on the Reactor Building net free volume. Therefore the replacement of thermal insulation will not have an adverse impact on the Reactor Building as a fission product barrier.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☒ No

BASIS:

ER-ANO-2002-1078-016 will install new MRI on the ROTSGs and connected piping, replacing existing MRI, calcium silicate and fiberous insulation. As discussed in response to Question 1, seismic and stress analyses were performed to demonstrate that the replacement MRI does not over-stress the connected piping or the RCS and does not come loose from the OTSG during a seismic event. The seismic analysis of the replacement MRI that was performed by Transco does not change an analysis or methodology that is currently discussed in the FSAR. The replacement of the non-metallic and MRI on the OTSGs and connected piping does not involve the use of new methodologies or methods of evaluation not described in the FSAR.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-026

I. OVERVIEW / SIGNATURES

Facility: ANO - Unit 1

Document Reviewed: ER-ANO-2002-1078-018, ANO-1 SG/RVCH Replacement - Interference Removal and Replacement

Change/Rev.: ERCN-3

System Designator(s)/Description:

Description of Proposed Change:

ER-ANO-2002-1078-018, ANO-1 SG/RVCH Replacement Project – Interference Removal and Replacement, provides for the removal of miscellaneous mechanical, electrical, and structural interferences that are not included in other Project ERs. Removal of these miscellaneous interferences is needed to facilitate the removal and replacement of the ANO-1 Once Through Steam Generators (OTSGs) and Reactor Vessel Closure Head (RVCH). Most of the removed interferences will be re-installed to their design configuration. Removal of two sections of Reactor Building (RB) facade and the relocation of the Reactor Building Purge system exhaust duct are the only two permanent plant changes. ER-ANO-2002-1078-018 also provides the basis for for operating the Reactor Building Breathing Air, Reactor Building Cooling System fans and coolers, and the Reactor Building Purge exhaust system with each of these systems in a configuration other that the designed configuration and to restore each of these systems when the altered configuration is no longer required.

Miscellaneous Mechanical Interferences

The interferences discussed below are inside and outside the Unit 1 Reactor Building. The interferences inside the Reactor Building are: 1) Reactor Coolant Pump (RCP) P32D Intermediate Cooling Water (ICW) System Supply pipe, 2) Breathing Air piping, 3) RCP Oil Collection piping, 4) Reactor Building Cooling System ductwork, 5) RCP P32B Drain Line Support, 6) Steam Generator E24A Hot Leg sample line and 7) Steam Generator E24B PT piping. The interferences outside the Reactor Building are: 1) Penetration Room Ventilation System (PRVS) ancillaries, 2) Emergency Feedwater Initiation and Control (EFIC) System Steam Line 3/8" Instrumentation Tubing, 3) Reactor Building Purge (RBP) system exhaust duct located at azimuth 44°, 4) the RB façade at 30° and 270°, and 5) miscellaneous structural components.

Reactor Coolant Pump (RCP) P32D Intermediate Cooling Water (ICW) System Supply Pipe

Inside the Reactor Building in the area of P32D approximately 20' of pipe JBD-8-3" will be removed. This ICW piping provides cooling water to Backstop Lube Oil Cooler E-195D for the P32D lube oil system. This pipe section will be disconnected at a flange on one end and cut on the other. Each of the four openings of the pipe ends will be temporarily covered to prevent foreign material intrusion.

Breathing Air Piping

The Breathing Air system provides a reliable supply of dry, oil-free, high quality compressed air for use with inline
respirators throughout the plant. Portions of the Breathing Air system piping that will be temporarily removed are located inside the "D" ring of the E24A Steam Generator area. Approximately 5' of JDD-6-1" and one pipe support will be removed, stored and reinstalled. After removal of the pipe section, a pipe cap will be temporarily installed on the pipe stub at Floor Elevation 424' to allow for operation of the Breathing Air system as required during the SGR outage. The temporarily installed pipe cap will be capable of withstanding the system normal design pressure of 125 psig.

This system may be operated in its temporary configuration as required to support other operations and construction activities. Restoration will be accomplished when the design configuration of the pipe no longer interferes with the replacement of the steam generator. The Breathing Air system is non-safety related and there is no impact on any SSC by operating this system in a temporary configuration.

ER-ANO-2002-1078-018 and this 50.59 Evaluation provide the necessary justification and instructions for operating the system in its temporary configuration and then restoring the system to its design configuration.

Reactor Coolant Pump (RCP) Oil Collection

The upper and lower bearing housings are provided with an oil retention barrier. Oil that may leak during RCP operation is collected in the upper and lower retention barrier which drains to a collection tank. The two collection tanks (T-90 and T-91) are located in the Reactor Building on elevation 336'-6". Oil collected from P32A and P32B goes to T-91 and P32C and P32D goes to T-90.

Reinstallation of the RCP oil piping and associated supports will be to the ANO Unit 1 design configuration. No change is made to the pipe routing or size.

Reactor Building Cooling System

The Reactor Building Cooling System (RBCS) is safety related. The RBCS functions to provide cooling to the Reactor Building during power operation, plant shutdown and accident conditions.

Portions of the Reactor Building Cooling System return air ductwork and supports which interfere with the temporary Reactor Building construction opening (ER-ANO-2002-1078-007, Reactor Building Opening) will be removed. The duct and supports are non-safety related, however, they are Seismic Category II/I. The Seismic qualification of the duct which remains installed will not be impacted by removal of a portion of the duct because the duct is supported in sections by individual independent supports. The duct removal and replacement must be sequenced according to the requirements for the timing of the removal and replacement of the concrete and liner plate. The duct must be removed from the liner before the concrete removal process has reached a depth of 30" (i.e., remaining concrete depth ≥ 15"). The duct must not be supported by the liner plate during reinstallation until the cure of the reinstalled concrete has obtained a compressive strength of 3000 psi as required by ER-ANO-2002-1078-007, Section 4.2.2.

The RBCS will be operated during all modes including defueled to provide an environment in the Reactor Building which supports personnel in the performance of their activities. The RBCS must also be available to remove decay heat as required by the 1R19 SOPP.

The RBCS will be temporarily operated in an altered configuration to support the environmental conditioning inside the Reactor Building during activities associated with RVCH and OTSG replacement. DRNs are included for P&ID M-261 which show the system configuration with the duct removed. Calculation CALC-ANO-ER-04-033, "Evaluation of HVAC RBCS in the Interim Configuration for the Unit 1 SGRP", Revision 0 provides the allowance to operate the system in the temporary configuration with a portion of the return air duct removed. The unit housing doors may be secured opened as required to provide an air flow path through the chilled water cooling coils and fans.

In order to support Reactor Building cooling and the work required on the return air duct, at least one cooler must be operated with the inlet plenum door secured in the open position to allow air to flow across the chilled water coils but by-pass the return air duct. Other fans should be available as required by the 1R19 SOPP. Construction media roll type filters (or equivalent) will be temporarily installed either at the duct opening or the filter frame in the air handling units upstream of the chilled water coils to prevent particulate from entering the chilled water and service water coils for operation during the construction activities of the outage. These filters will be periodically checked and replaced as required to ensure cleanliness and prevent excessive pressure drop on the fans.

The portion of the return air duct which will be temporarily removed is located between elevations 399'-6" and 426'-6" at azimuth 270°. The duct being removed is all welded construction with angle reinforcement. The cut areas of the duct sheet metal will be seal welded when reinstalled. The support angle will be cut from the baseplates which are attached to the Reactor Building liner plate. Upon reinstallation, the angle will be rewelded to the baseplates. No welding to the Reactor Building liner plate is required.

The Reactor Building Cooling ductwork and its associated supports will be reinstalled to ANO Unit 1 design configuration. Reinstallation may begin when the ductwork is no longer an interference and must be complete prior to any SOPP requirements for refueling and entering Mode 4. However, transfer of the ductwork static weight load to the liner plate will not occur until construction opening repairs meet strength requirements defined in ER-ANO-2002-1078-007. Reinstallation of permanent duct, supports and stiffeners may be accomplished with the fans operating provided that there exists no detrimental consequence to the welding procedure (vibration or air flow/negative pressure on sheet metal duct) or to personnel safety.

ER-ANO-2002-1078-018 and this 50.59 Evaluation provide the necessary justification and instructions for operating the system in its temporary configuration and then restoring the system to its design configuration.

Penetration Room Ventilation System (PRVS) 12" Exhaust Pipe Ancillaries

The Penetration Room Ventilation System (PRVS) filters air from the penetration areas in the event of penetration leakage from the Reactor Building during a loss of coolant accident. The PRVS is Seismic Category 1 and safety related. The PRVS initiates on an Engineered Safeguards Actuation System (ESAS) signal.

Technical Specification 3.7.11 requires the PRVS be operable in Modes 1, 2, 3, and 4. The PRVS is not required to be operable in Modes 5 and 6 and when the reactor is defueled. In order for the PRVS to be operable, among other requirements, Tech Spec Bases B3.7.11 c. requires that the ductwork be operable for the PRVS to be operable.

The PRVS will be affected by two areas of work: 1) the tubing for flow element FE-9836 will be temporarily removed and then reinstalled, and 2) the tubing for Isokinetic Probe AE-9835 will be temporarily removed and reinstalled. The insulation and heat tracing for the above sample tubing will be removed and reinstalled as required to support tubing removal and reinstallation.

All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled. All of the tubing, supports and insulation will be reinstalled per the design configuration prior to entering Mode 4. Replacement parts conforming to existing design specification or equivalent will be used as required. Compression fittings will be used for reinstallation of tubing.

Steam Generator E24A Hot Leg Sample Line

Approximately 17' of CCA-13-3/4" in the Reactor Building area around E24A will be temporarily removed along with 4 supports. This sample line is off of the Reactor Coolant Piping hot leg inlet to the steam generator. The pipe will be cut on one end and a fitting weld removed on the other. The supports will be removed by unbolting the intermediate connection plates. One snubber and two spring cans will be removed. Most hardware will be reused. Replacement bolt material conforming to or equivalent to design specification will be used if needed. The removed piping and the piping that remains will be closed on the ends to prevent foreign material/debris intrusion. FME and cleanliness controls will be per SGT Quality Execution Procedure 10.04, General Housekeeping, Cleanliness, and Foreign Material Exclusion (FME) Requirements. All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled.

The removal of this pipe will also provide a vent path for fill activities while the FME barrier is still in place in the Hot Leg piping. Partial refilling of RCS is required to begin refueling operations, therefore the FME barrier must remain until the hot leg elbow replacement welding is finished. To support RCS refilling operations, a temporary vent path must be provided to release the air displaced during liquid refill of the vessel. The open ended sample line (covered with an FME screen) will provide that vent path.

Reinstallation will be per the design configuration except that an additional fitting may be used to connect the pipe where it was cut. The system will be restored prior to the plant entering Mode 45.

Steam Generator E24B Hot Leg Pressure Transmitter PT-1039 Line

An elbow on 1" pipe CCA-1-1" will be removed to provide a vent path for fill activities while the FME dams are still in place in the Hot Leg pipe. Partial refilling of RCS is required to begin refueling operations, therefore the FME barrier must remain until the hot leg elbow replacement welding is completed. Therefore, a temporary vent path must be provided to release the air displaced during liquid refill of the vessel. The open ended PT line (covered with an FME screen) will provide that vent path.

Reinstallation will be per the design configuration. The system will be restored prior to the plant entering Mode 5.

Steam Line 3/8" Instrumentation Tubing

The Emergency Feedwater Initiation and Control (EFIC) System (formerly - Steam Line Break Instrumentation and Control (SLBIC) System) is designed to protect against the consequences of a simultaneous blowdown of both steam generators. One section from each of the four 3/8" tubing lines will be temporarily removed in the area of the Reactor Building exterior buttress at azimuth 60°. All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 4.

RCP P32B Inlet Drain Line Support

The drain line support H002 is temporarily removed to eliminate an interference with SG E24B RCS cold leg temporary support. The RCS temporary support is installed and removed by ER-ANO-2002-1078-015. Support H002 will be disconnected from the RB floor by removing the nuts and washers from the anchors. Also the support will be disconnected from piping CCA-13-1½" by unbolting the pipe clamp from the sway strut rod end. The pipe clamp will remain attached to piping. All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 4.

Reactor Building Purge Exhaust Duct and Isokinetic Probe (AE-9820)

The Reactor Building Purge system serves to provide clean air in the Reactor Building during Modes 5, 6 and Defueled but cannot be operated during Modes 1 – 4 due to the inability of the Reactor Building isolation valves to close against DBA LOCA pressures. The exhaust air is sampled for radiological activity via an isokinetic probe located in the duct sections being relocated. Tubing connected to the probe is insulated and heat traced to prevent condensation of the air being transported through the tubing to the SPING system. Air is returned through tubing to the exhaust air duct upstream of the ER affected portion of the duct.

To accommodate SGRP tendon work, existing purge exhaust duct from fan VEF-15, located outside Unit 1 Reactor Building will be relocated from azimuth 44 to azimuth 30. This vertical section of ductwork has sample probe AE-9820 mounted in it at approximate elevation 482'-0". Probe AE-9820 is connected to ERGE-RMS (extended range gaseous effluent radiation monitoring system) monitor RX-9820 via ¾" tubing. This tubing is heat traced (refer to Electrical portion of ER-ANO-2002-1078-018 for details of heat trace removal). This heat tracing, along with the associated tubing and probe (AE-9820) will be relocated with the vertical ductwork. Insulation will be removed from sample probe AE-9820 back to and including the section of tubing to be removed. Heat tracing will be removed from sample probe AE-9820 back to and including the section of tubing to be removed. Heat trace cable will be rolled up and tied back out of the construction activity area. After the section of tubing that was removed has been re-installed, heat tracing and insulation will be re-installed.

Conduit EC2251 containing cables associated with the Reactor Building isolation valve CV-7401 will interfere with the relocated ductwork. This conduit needs to be removed, adjusted and reinstalled so that the conduit will not extend past 3" from the outside Reactor Building wall. This work is performed in the Electrical portion of ERANO-2002-1078-018.

Steel base plates will be attached to the RB wall in the location where the exhaust duct will be moved to during the outage. This activity will be performed prior to the outage with Unit 1 expected to be in Mode 1. This activity is addressed in a separate 10 CFR 50.65 Assessment.

All of the above purge system work, except baseplate installation, will be performed after the initial Reactor Building purge and Reactor defueling activities which require the purge system to be operable. Staging of materials and insulation removal may be performed during any mode. No work which breaches the system will be performed prior to defuel and the work will be completed prior to refuel.

ER-ANO-2002-1078-018 and this 50.59 Evaluation provide justification for placing the system in service in its temporary configuration and then restore the system to its design configuration. ER-ANO-2002-1078-018 provides the instructions necessary to allow the purge system to be operated during the time when the vertical duct flute and radiation monitor are disconnected. The temporary configuration will provide an opening in the duct above the Auxiliary Building roof for air to discharge. This air will be manually sampled for radiological contaminants and the system secured if any are present.

Total air flow will be assumed not to change sufficiently to impact off-site dose calculations due to the removal of a portion of the exhaust duct.

Façade

The façades at azimuths 30° and 270° being removed are painted sheet metal and steel structures which have no current system or plant function other than the aesthetics of the Reactor Building exterior. The façades are to be removed and disposed of as directed by ANO. This activity will be performed prior to the outage with Unit 1 expected to be in Mode 1 but may take place in any plant mode. This activity is addressed in a separate 10CFR50.65 Assessment.

Miscellaneous Electrical Interferences

RBCS Ductwork EL. 401'-6"

As previously discussed, the ductwork at 401'-6" elevation above the existing equipment hatch will be removed. The following conduits, temperature elements, and dew point elements are connected directly to the ductwork that is to be removed: Conduit J5379 (Cables I508B and I508D) feeds to HE-6278, Conduit J5380 (Cable I529D) feeds to TE-6278, Conduit J5381 (Cables I509B and I509D) feeds to HE-6279 and Conduit J5382 (Cable I530D) feeds to TE-6279. The cables will be de-terminated at the instrument and the associated cable will be pulled back and all conduits will be removed back to the first available junction box or cable tray that provides sufficient clearance for removal of the ductwork. The instruments will be removed and stored for reinstallation once the ductwork has been reinstalled. All of the above listed components are not safety related.

All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 4.

E24A Platform EL 424'-6" (Crow's nest)

To support the removal of the E24A platform steel and the RCS Hot Leg High Point Vent piping, four conduits that run along the structural steel and will be removed back to the fittings (LB condulets) at the north cavity wall. The conduits are oriented on the structural steel from top conduit to bottom conduit – SC2048, SC1079, SR2025 and J5241. Also conduits SC2046, SC2047, SC1077 and SC1078 and cables associated with the Valves SV-1082, SV-1084, SV-1081 and SV-1085 respectively are removed from the devices back to the LB condulet on the D-ring.

There is also a cable that is attached to building steel that is connected to TE-1080. There is an accessible plug near TE-1080. The cable will be unplugged and removed back to the D-ring and rolled up and tied back out of the rigging path.

All work on this system will be performed in Mode 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 5. The RCS Hot Leg High Point Vent piping will be removed and replaced by ER-ANO-2002-1078-015.

Jib Cranes (EL. 424'-6")

There are four disconnect switches with associated receptacles for jib cranes. The disconnect switches are S-101, S-102, S-103 and S-104. Only the conduits and associated cables for the disconnect switches (S-102, S-103, S-104) are required to be removed. The associated hoists will not be installed on these jib cranes. Switch S-101 is not affected by the ER.

All work on this system will be performed in any plant mode or while the reactor is defueled and will be restored prior to the plant entering Mode 4.

E24A Vibration Detection Element (VBE) EL. 392'-7"

Vibration and Loose Parts Monitoring System accelerometers are located on steam generator E24A. ANO-1 I&C Shop Technicians will perform the removal and reinstallation of the devices.

All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 4.

E24B Platform EL 424'-6" (Crow's nest)

To support the removal of the E24B platform steel and the RCS Hot Leg High Point Vent piping, four conduits that run along the structural steel and have to be removed back to the fittings (LB condulets) at the south north cavity wall. The conduits are, SC2042, SC1073, SR1021 and J5239. Also conduits SC2040, SC2041, SC1071 and SC1072 and cables associated with the Valves SV-1092, SV-1094, SV-1091 and SV-1093 respectively are removed from the devices back to LB condulet on the D-ring.

There is also a cable that is attached to building steel that is connected to TE-1090. There is an accessible plug near TE-1090. The cable will be unplugged and removed back to the D-ring and rolled up and tied back out of the rigging path.

All work on this system will be performed in Mode 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 5. The RCS Hot Leg High Point Vent piping will be removed and replaced by ER-ANO-2002-1078-015.

E24B Vibration Detection Element (VBE) EL. 392'-7"

Vibration and Loose Parts Monitoring accelerometers are located on steam generator E24B. ANO-1 I & C Shop Technicians will perform the removal and reinstallations of the devices.

Cable I332B will be de-terminated at VBE-1039B and pulled back through conduit J5215 to pull box TB 392. The cable will be rolled up and tied back out of construction activity. Conduit J5215 will be removed.

Cable I332F will be de-terminated at VBE-1039A and pulled back through conduit J5251 to pull box TB 392. The cable will be rolled up and tied back out of construction activity. Conduit J5251 will be removed.

All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 4.

<u>Auxiliary Building Area 3 & 4 EL. 404'-0"</u>

To support Reactor Building tendon activities, the following five conduits and associated cables in Auxiliary Building Areas 3 and 4 will be removed:

B5114 and B5116 - RB Tendon Gallery Recirculation Fan

EC2251 – RB Purge Outlet valve

EC2813 – EFW Pump Turbine steam admission

EJ2026 – Steam generator E24B Pressure Control

All work on these systems will be performed in Modes 5 and 6 or while the reactor is defueled. The systems will be restored prior to the plant entering Mode 4.

Reactor Building Tendon Gallery Supply Fan VSFM 23C, VSFM 23D

Cable B4151A (for fan VSFM 23C) and cable B4152A (for fan VSFM 23D) will be de-terminated at the fan motor. Fans VSFM 23C and VSFN 23D will be removed and stored for later reinstallation. There are no mode restrictions for these activities.

Miscellaneous Electrical Interferences E24A and E24B

To ensure protection of plant equipment, ANO-1 I&C Shop Technicians will perform the removal and reinstallation of SG E24A RTDs TE-2661, TE-2662, TE-2664/2665 (dual element RTD), SG E24B RTDs TE-2611, TE-2612, TE-2614/2615 (dual element RTD) and SG E24A shell thermocouples TE-2654, TE-2655,TE-2656, TE-2657 and TE-2658, SG E24B shell thermocouplesTE-2604, TE-2605, TE-2606, TE-2607 and TE-2608. The work will include the removal/determination and pull back of wiring prior to the removal of the Original OTSGs, as well as re-installation and re-termination of these devices after the replacement of OTSGs work has been completed. ANO-1 I&C will provide for safe storage of these components until the outage has progressed to the point where they can be reinstalled. ANO-1 I&C Shop Technicians will perform the work.

Ground straps embedded in the concrete base pedestal are connected to the skirt of the OTSGs (E24A and E24B). The ground straps are bolted to the skirt. The ground straps will be disconnected from the Original OTSG's skirts and will be reconnected to the replacement OTSG's skirts.

On the north side of the OTSG E24A TOS EL 392'-11½", a # 4/0 AWG ground conductor is clamped to the web of the east-west W14X78 beam. The beam is going to be removed as a structural interference. The ground conductor will be removed from the beam clamps and split screw splice connections and secured to building steel on the north wall. This ground will be replaced after the steel is replaced.

Miscellaneous Electrical Interferences Outside the Reactor Building

The Super Particulate Iodine Noble Gas (SPING) monitor tubing for AE-9835 and the associated heat tracing will be temporarily removed and reinstalled at approximate elevation 437'-0" as part of the mechanical portion of this ER. The tubing will interfere with the tendon removal/replacement equipment. Probe AE-9835 is connected to monitor RX-9835 via this ¾" tubing. Electrically, the heat tracing system for probe AE-9835 consists of one circuit (circuit no.13) and two (2) thermocouples.

The Heat Tracing associated with the EFIC System tubing MS-54, MS-56, MS-58 and MS-60 is required to be removed temporarily to facilitate mechanical work and to be reinstalled after the completion of the mechanical work.

All work on this system will be performed in Modes 5 and 6 or while the reactor is defueled. The system will be restored prior to the plant entering Mode 4.

Miscellaneous Structural Interferences

Steam Generator Cavity Platform Removal

A motion study was used as the basis for determining what structural components are considered interferences within the steam generator cavities. These interferences were confirmed in a pre-replacement outage walkdown. Based on the motion study and walkdown, the cavity platforms at approximate elevations 393 ft and 384 ft will be partially disassembled to provide the required clearances. The remaining platforms at approximate elevations 347 ft and 376 ft clear the spatial envelope and do not require modification. Grating on the affected portions of the platforms will be tagged and removed until the support steel is reinstalled. The portions of grating remaining will be qualified for 100 psf live load. No shielding loads are permitted on the structural steel in steam generator cavities at approximate elevations 376', 384', and 393' during the duration of the modified configuration without engineering approval. The temporary platform configurations are qualified to ANO Seismic Category II/I criteria, and are therefore acceptable for plant operating Modes 5 and 6 and while the reactor is defueled.

Steam Generator Cavity Platform Reinstallation

Reinstallation of the platforms is in accordance with plant requirements including the AISC Manual of Steel Construction 6th Edition and the AWS Structural Welding Code, AWS D1.1, 1992. All structural bolted joints that are disassembled use new bolting material for reinstallation of the removed steel. Structural steel members that are cut for removal are welded back using full penetration welds. Therefore, the platforms are reassembled to their design condition. There are no permanent changes to the platforms other than the structural steel welding. All structures will be restored prior to entering Mode 4.

Elevation 424'-6" Steam Generator Cavity Decking Steel and Jib Cranes

The associated floor grating and two of the 33WF floor beams in each steam generator cavity will be removed. Three of the four RCP jib cranes (L32B, L32C, & L32D) are attached to the 33WFs and will be removed to provide clearance for the steam generator replacements. The jib cranes will be partially disassembled by removing the jib arms. The RCS Hot Leg High Point Vent system is mounted below the two 33WF beams along with the platform at elevation 419'-7½". The jib cranes, the 33WFs, the platform at elevation 419'-7½", and the RCS Hot Leg High Point Vent piping will all be removed as one unit. In order to accomplish this, W4 beams will be attached to the 33WFs to assist in the rigidity of the whole unit. After removal, the steel is stored in a reserved location on the 424'-6" elevation. One W10 beam that spans from the 419'-7½" platform framing to the cavity wall will be cut near the wall to provide a means for its removal. This W10 beam will be temporarily supported by the temporary W4 beam. The qualification of the entire unit is included with the structural interferences calculation (CALC ANO-ER-04-031). The 33WF beams are removed by unbolting. Reinstallation of these beams requires bolting the 33WFs using new bolts, welding the W10 with full penetration welds, removal of the W4s, and reassembling the jib cranes. Upon reassembly, the structures will be restored to their design condition which will occur prior to entering Mode 4.

Removal and reinstallation of the 33WFs will be coordinated with ER-ANO-2002-1078-015 which provides instructions for the removal and replacement of the RCS Hot Leg High Point Vent system. Also, temporary handrails must be provided as required for personnel safety at locations where the floor grating has been removed.

Elevation 401'-6" 12WF Floor Beam

The 12WF floor beam located parallel to the west side of the reactor building equipment hatch will be removed along with its associated grating to provide additional access for removal of the Reactor Building Cooling System air return duct. Removal of the beam provides approximately 2 feet of additional horizontal movement for the bottom of this duct after the duct is cut. The duct will be cut and removed as part of the mechanical portion of the ER. The 12WF beam will be removed and reinstalled by bolted connections. New bolting is used for the reinstallation. The beam will be restored to its design condition and the grating replaced prior to entering Mode 4. No qualification is required for the removal of this beam since its primary function is to resist floor loadings and its contribution to the overall stability and strength of the reactor building structural steel is insignificant.

Elevation 430'-6" Purge Valve Access Platform

The elevation 430'-6" access platform for the Reactor Building purge exhaust duct and valve CV-7403 will be partially disassembled to provide additional laydown area for the steam generator cavity decking floor beams. Approximately 3 feet will be removed from the outside face of this platform to provide the required space. The platform will be removed by unbolting and cutting two W8 beams. Reinstallation of the platform requires new bolting for the reassembled joints and welding the W8 beams with full penetration welds. Thus, upon reassembly, the platform is restored to its design condition. Barriers will be provided in the field to limit access to this platform during the time it is disassembled. No qualification is required for the partial removal of this platform since the vertical bracing and structural support points remain intact. The structure will be restored prior to entering Mode 4.

ORVCH Steel Removal

The structural interferences are removed from the ORVCH in the reactor building as required for rigging and handling, transport, and storage spatial requirements. The items removed for disposal include:

- The control rod drive mechanism (CRDM) cooling duct. This includes all the duct that normally remains attached to the service structure for movement to the head stand.
- The CRDM cooling duct supports. These supports are attached to the service structure steel, and are cut back to allow for the spatial requirements.
- The reactor head high point vent pipe supports, piping, and valves. These items are mounted on the service structure platform, and are cut back close to the top of the platform decking. Note that the PT-1070 instrument and its support, which are part of this system, are salvaged for reuse by ER-ANO-2002-1078-0639-000.
- The service structure platform handrail. The handrail is removed just above the toe plate.
- The ORVCH lifting pendants. The lifting pendants and associated hardware shall be carefully removed and stored as a contingency for any problems experienced with the new lift pendants and associated parts.

The ORVCH steel removal will be coordinated with ER-ANO-2002-1078-017 which addresses the radiation control measures needed to prepare the ORVCH for movement to the OSGSF. The remaining steel on the ORVCH cannot have sharp edges or projections that could cause damage to coverings that are to be used for contamination control.

The ORVCH steel removal must also be coordinated with ER-ANO-2002-1078-011 which addresses the design, temporary installation, and use of construction aid equipment to perform rigging and handling activities inside the reactor building. SGT will install a special lifting device for the ORVCH which will be used in the downending process required to move the ORVCH through the reactor building wall construction opening. In addition to ensuring that clearances are provided for the lifting device itself, the spreader beam used as part of the ORVCH lift rigging must pass over the service structure platform during the downending process, and requires clearance above the platform decking.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-026)**	**Sections I, II, and IV required**

Preparer: Wayne R. Wasser / **ORIGINAL SIGNED BY WAYNE WASSER** / Adecco Tech / SG-RVCH / 6-23-05
Name (print) / Signature / Company / Department / Date

Reviewer: Doyle G. Adams / **ORIGINAL SIGNED BY DOYLE ADAMS** / EOI / SG-RVCH / 6-23-05
Name (print) / Signature / Company / Department / Date

OSRC: J.N. Miller / **ORIGINAL SIGNED BY J.N. MILLER** / 6-27-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	ANO-1 FSAR Figure 9-14 (P&ID M-218 sheet 5), ANO-1 FSAR Figure 6-13 (P&ID M-261, sheet 1), and ANO-1 FSAR Figure 5-7 (P&ID M-261), sheets 1 & 2) will be temporarily affected to allow for system operation in an altered configuration but not permanently changed.
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☐ **Yes**

☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change <u>AND</u> initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. <u>Basis</u>

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

In support of replacement of the Unit 1 steam generators and RVCH, a number of mechanical, electrical, and structural interferences will be moved to eliminate the interference. Each interference will be removed in a Mode when it is not required to be in service. Each interference will be returned to its design configuration with the exception of two sections of the RB façade which will not be replaced.

<u>Operating License/Technical Specifications (TSs)</u>

Of the mechanical/electrical/ structural interferences that will be removed and later replaced, only the Reactor Building Purge System, Reactor Building Cooling System, Penetration Room Ventilation, and EFIC Systems are discussed in the OL/Tech Specs. The Intermediate Cooling Water, Breathing Air, SG Hot Leg Sample, cold leg drain, Jib Cranes, Vibration and Loose Parts Monitoring System, ground strap, CRDM Cooling, shielding monorail, Reactor Building façade, Reactor Building tendon gallery fans, Super Particulate Iodine Noble Gas monitor, RCP Oil Collection System, and Reactor Building Platforms are not discussed in the Operating License or Technical Specifications. They are not discussed in the Tech Spec Bases, Technical Requirements Manual, or Core Operating Limits Report.

The Reactor Building Purge System is discussed in Tech Spec Sections 3.6.3 and 3.9.3; TS Bases Section 3.6.3 and 3.9.3; and TRM Section 3.6.1. No changes to the Tech Specs, Tech Spec Bases, or TRM are needed.

The Reactor Building Cooling System is discussed in Tech Spec Sections 3.3.6 and 3.6.5; TS Bases Section 3.3.5, 3.3.6, 3.3.7, 3.6.5, and 3.7.7; and TRM Section 3.6.5. No changes to the Tech Specs, Tech Spec Bases, or TRM are needed.

The Penetration Room Ventilation System is discussed in Tech Spec Sections 3.7.11 and 5.5.11 and in Tech Spec Bases Sections 3.3.5, 3.3.6, 3.3.7, and 3.7.11. No changes to the Tech Specs or Tech Spec Bases are needed.

The Emergency Feedwater Initiation and Control (EFIC) System is discussed in Tech Spec Sections 3.3.11, 3.3.12, 3.3.13, 3.3.14 and, 3.8.9. It is discussed in Tech Spec Bases Section 3.3.5, 3.3.6, 3.3.7, 3.3.11, 3.3.12, 3.3.13, 3.3.14, 3.7.2, 3.7.3, 3.7.5, 3.8.7, and 3.8.9. No changes to the Tech Specs or Tech Spec Bases are needed.

TRM 3.4.3 limits steam generator secondary side pressure to <200 psig with the steam generator shell temperature <100°F. The steam generator shell thermocouples are used to show compliance with this requirement. With the secondary side of the steam generator piping cut for steam generator replacement, pressurization of the secondary side of the steam generators to >200 psig is not possible. The thermocouples will be returned to an operable status prior to pressurization of the ROTSGs. No changes to the TRM are needed. The SG Downcomer and Downcomer Inlet RTDs are not discussed in the Tech Specs, Tech Spec Bases, or TRM.

The RCS high point vent is discussed in TRM Section 3.4.2. No changes are required.

FSAR

The following systems that are impacted by this ER are discussed in the FSAR. The function of each system remains unchanged therefore no changes to the FSAR are needed.

The Intermediate Cooling Water (ICW) System is discussed in various locations in the FSAR. No changes to these sections are needed. This portion of the ICW System is not in service during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

The Breathing Air (BA) System is discussed in various locations in the FSAR. No permanent change to the FSAR is needed. FSAR Figure 9-14 shows the BA System. During the SGR outage, a section of the BA System piping will be removed and the remaining piping will be capped as part of ER-ANO-2002-1078-018 so that the BA System can be returned to service. ER-ANO-2002-1078-018 includes temporary changes to FSAR Figure 9-14. Prior to completion of the SGR outage, the BA System will be returned to its design condition. Changes to this system are temporary and the system will be restored to design condition.

The RCP Oil Collection System is discussed in various FSAR Sections. No changes to these sections are needed. The RCP Oil Collection System is not in service during the SGR Outage. Changes to this system are temporary and the system will be restored to design condition.

The Reactor Building platforms are discussed in FSAR Section 5.1.7.1. No change to this section is needed. Changes to the platforms are temporary and will be restored to their design condition.

The Reactor Building Cooling System (RBCS) is discussed in a number of locations in the FSAR. These sections describe the operation of the RBCS during normal (power) operation and in post-accident conditions. The planned temporary changes to the RBCS will not affect the operational modes that are described in the FSAR. ER-ANO-2002-1078-018 includes temporary changes to FSAR Figure 5-7. Prior to completion of the SGR outage, the RBCS will be returned to its design condition. Changes to this system are temporary and the system will be restored to design condition.

The Reactor Building Purge System is discussed numerous locations in the FSAR. No changes to these sections are needed. The planned temporary changes to the RBP will not affect the operational modes that are described in the FSAR. ER-ANO-2002-1078-018 includes temporary changes to FSAR Figure 5-7. Prior to completion of the SGR outage, the RBP will be returned to its design condition. Changes to this system are temporary and the system will be restored to design condition.

The RCS hot leg sample is discussed in FSAR Section 5.2.2.4.1. No changes to this section are needed. The RCS hot leg sample is not in service during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

The Penetration Room Ventilation System (PRVS) is discussed in various locations in the FSAR. No changes to these parts of the FSAR are required. The PRVS is not required to be operable during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

The Emergency Feedwater Initiation and Control (EFIC) System is discussed in various FSAR Sections. No changes to these sections are needed. EFIC is not required to be operable during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

The Jib Crane is discussed in FSAR Section 9.6.1.7.1. No changes to the FSAR are needed. Changes to the Jib Cranes are temporary and will be restored to their design condition.

The RCS high point vent is discussed in various parts of the FSAR. No changes to the FSAR are needed. The high point vent is not required to be operable during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

The Vibration and Loose Parts Monitoring System are discussed in various parts of the FSAR. No changes to the FSAR are needed. The Vibration and Loose Parts Monitoring System is not required to be operable during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

The Super Particulate Iodine Noble Gas (SPING) Monitor System is discussed in various parts of the FSAR. The SPING Monitor System is not required to be operable during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

Steam Generator Shell Temperature indication is discussed in various parts of the FSAR. No changes to these sections are needed. The Steam Generator Shell Temperature indication is not required to be operable during the SGR outage. Changes to this system are temporary and the system will be restored to design condition.

Test or Experiment Consideration

Removal and reinstallation of mechanical, electrical, and structural interferences in support of steam generator and RVCH replacement will remove a number of systems from service. Each of these systems with the exception of two sections of the RB façade, will be returned to its design or modified configuration. Each system that is removed from service will be appropriately tested prior to return to service. None of the systems returned to service will be used for functions that are outside of their design basis. Removal and reinstallation of these interferences is not considered a test or experiment.

Platforms, handrails, steel beams, and grating in specific locations, along with the jib cranes located at elevation 424 (nominal), have been identified as interferences inside the Reactor Building. These components will be removed and later returned to their design configuration. Removal of these structures is not considered a test or experiment.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search: LRS 50.59 – Unit 1	Keywords: “intermediate cooling water”, ICW, “breathing air”, “oil collection”, “Containment Cooling”, “Reactor Building Purge”, “Reactor Building Cooling”, “RB Cooling”, “reactor coolant system” w/20 sample, platform*, grating*, “penetration room ventilation”, PRVS, “Emergency Feedwater Initiation”, EFIC*, jib, “loose parts”, ground w/20 strap*, “steam generator” w/20 ground, CRD* w/5 cooling, “control rod” w/5 cooling, “high point”, monorail, “hydrogen purge”, “super particulate”, SPING*, “temperature sensor”, “temperature sensors”, “temperature element”, “temperature elements”, “temperature indicator”, “temperature indicators”, “pressure indicator”, “pressure indicators”, “flow element”, “flow elements”, L32*, L-32*, drain w/10 (pip* OR line*), “shell temperature”, “inlet temperature”, vibration w/20 detect*”, “tube to shell”, tube*to*shell, “SPDS”, “Safety Parameter Display”, ICS, “Integrated Control”, “vibration and loose parts”, “green train”, “red train”, HE-6*, E-33*, TE-6*, SV-108*, SV-109*, GCR*, RCR*, PT-10*, E-4*, E-252*, VBE*, TE-2*, RJI*, S-10*, TE-2*, E-7*, JB*, J53*, NJ*, SC1*, SC2*, GJI*, B532*, B44*, J52*, SC*1*, SC*2*, deck*, “building internals”, “nitrogen system”, “nitrogen supply”, thermocouple* w/20 (CRD OR control), (plate* OR clamp*) w/20 (CRD OR control), position indicator*, (duct* or “duct work”) w/20 (reactor OR containment), lub* near/10 (RCP OR “coolant pump”), SLBIC, “steam line break instrumentation”, M-*, ERG*, “extended range”, isokinetic, ae-*, VSFM*, facade, architectural, fascia, pendant*, TE-10*, E-195*, E195*, T-90, T90, T-91, T-91, “lube oil”w/5 cooler*, ZS*, “SV-7401”, “I/P-2618”, VEF-15*, VEF15*, “CV-7401”, EC2*, EJ2*, B511*, GC*, SV-10*, SR1*, crow*, VBE*, “tendon gallery”, hotleg, “hot leg”, “pressure transmitter” w/20 reactor
LBDs/Documents reviewed manually: ANO Unit 1 SAR	Chapter 7 and Sections 1.4.39, 4.2.2.5, 4.2.3.8, 4.2.4.7, 5.2.2.4.1, 5.2.4.4, 5.2.6, 5.2.7, 6.3, 6.5, 9.6.1.7.1, 6.6.2.1, 7.1.3.2.4, 9.6.1.1, 9.7,11.1.3.4.1, 11.1.3.8, 13.6, 14.2.2.5.2.3, Tables 6-10 and 6-13, and Figures 5-7, 6-4, 6-10, 6-13, 9-6, 9-7, 9-8, 9-10, 9-14, and 9-20.

5. **Is the validity of this Review dependent on any other change?** (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☐ **Yes**
☒ **No**

If "YES," list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☒	☐	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☒	☐	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

Access to the Auxiliary Building roof area at Elevation 358'-0" (area between the Auxiliary and Reactor Buildings) will be required. Access will be via roof hatch number 54 and door number 482. Door 482 is an entrance portal from the outside to the inside of the plant via the Auxiliary Building. To prohibit entrance to this area of the plant, the door is bolted. Compensatory measures will be required during the time that this door is unbolted. Additionally this portion of the Auxiliary Building roof area should be considered a "confined space". Ongoing activities in this area should be coordinated and communicated with ANO Safety, RP, and Security to ensure compliance with applicable requirements.

D INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The accidents that are discussed in the FSAR that are applicable in Modes 5 or 6 or are not mode related are: fuel loading errors (FSAR Section 14.1.2.10), fuel handling accident (FSAR Section 14.2.2.3), and waste gas tank rupture (FSAR Section 14.2.2.7). The fuel loading errors, fuel handling accidents, and waste gas tank rupture are unrelated to activities performed by ER-ANO-2002-1078-018. The activities in this ER (except removal of two sections of RB façade, installation of the RB purge base plates, and removal of the RB tendon gallery fans) occur during shutdown (Modes 5, 6, and defueled) and have no impact on the previously discussed FSAR accidents. Removal of the RB façade sections and tendon galley fans and installation of the RB Purge base plate will be performed outside of the Reactor Building and will have no impact on previously evaluated accidents. The RB façade and RB Purge base plate activities are addressed in a separate 10 CFR 50.65 Assessment.

   Therefore, the implementation activities of this ER will not increase the frequency of any accident previously evaluated in the FSAR where power or hot operation (Modes 1-4) is assumed.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   A number of the interferences that will be temporarily removed by this ER perform safety functions during normal operating modes. Work on these systems will not begin until the Unit is in a mode where the system is not required by the Tech Specs or TRM, as applicable, and the system has been released for work by EOI. Upon restoration, each SSC will be restored to its design configuration.

   Therefore, none of the planned activities will increase the occurrence of a malfunction of an SSC important to safety previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   ER-ANO-2002-1078-018 temporarily removes and replaces a number of mechanical, electrical, and structural interferences that function to mitigate the consequences of accidents previously evaluated in the FSAR. None of the SSCs that are part of this ER are accident initiators. The function of some of the components is to mitigate or assess accidents. These SSC will only be removed when they are not longer required to perform their safety related function. These SSCs will be restored to their design condition.

   Therefore, the removal and replacement of the interferences described in ER-ANO-2002-1078-018 does not result in any increase in the consequences of an accident previously evaluated in the FSAR. It can be further stated that there is no more than minimal increase in the dose expected from an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

With the exception of the RB façade removal, SSCs that are temporarily removed as part of ER-ANO-2002-1078-018 will be restored to their design or modified condition. This ER does not include any changes to the operating characteristics or functions of any affected SSCs. Likewise, there are no changes to the design basis for any of the affected SSCs. In essence, each SSC will be returned to an equivalent condition both physically and operationally.

Additionally, during Modes 5 and 6, partially disassembles SSCs will be left in a condition so as not to create a Seismic II/I condition to SSCs that are required for these modes.

Therefore, the removal and replacement of the interferences described in ER-ANO-2002-1078-018 does not result in any increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR. It can be further stated that there is no more than a minimal increase in dose expected from a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

ER-ANO-2002-1078-018 removes a number of mechanical, electrical, and structural interferences. Interference removal activities will not begin on each system until it is no longer required to be operable and is released by ANO-1 Operations. Each system will be returned to its design configuration with the exception of the RB façade. Two sections of the façade on the exterior of the Reactor Building will be removed and not replaced. The Reactor Building Purge exhaust duct will be relocated from azimuth 44° to azimuth 30° and will be installed to its original design conditions.

Therefore, the planned interference removals will not create the possibility for an accident of a different type than previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

ER-ANO-2002-1078-018 removes a number of mechanical, electrical, and structural interferences in support of the SG/RVCH replacement. All SSCs (except two sections of the RB facade) will be restored to their design or modified configuration. After restoration, all SSCs will remain qualified to perform their function in normal conditions and all applicable accident conditions. ER-ANO-2002-1078-018 does not change any design, system or functional parameters of each affected SSC.

Therefore, ER-ANO-2002-1078-018 will not create the possibility for a malfunction with a different result than previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

ANO-1 fission product barriers include fuel cladding, RCS Boundary, and the Reactor Building Pressure Boundary.

<u>Fuel Cladding</u>
The planned removal of interferences inside the Reactor Building will not begin until the Unit is in Modes 5, 6, or the Reactor has been defueled. The only activity that could be a concern for loose parts entering the reactor is the RCS hot leg sample line removal. FME practices and procedures will be followed to prevent the introduction of foreign material into the RCS.

RCS Boundary
The removal of the RCS hot leg sample and pressure transmitter lines are the only interferences that impact the RCS pressure boundary. The lines will be removed with the Unit in Mode 6 or with all fuel removed from the reactor vessel and when released by ANO-1 Operations. They will be returned to their design configuration prior to filling the RCS above the pipe cuts and prior to entry into Mode 5. All work on the system will be in accordance with ANSI B31.7. The integrity of the RCS pressure boundary will be verified by conducting an inservice leak test in accordance with ASME Section XI.

Reactor Building Pressure Boundary
The removal and replacement of SSCs describe in ER-ANO-2002-1078-018 will not adversely impact the pressure retaining capability of the Reactor Building or other fission product barriers. Supports for the Reactor Building Cooling System return ductwork will be welded to the existing base plates. No welds will be made to the Reactor Building Liner Plate.

Therefore, the activities describe in ER-ANO-2002-1078-018 do not affect the design basis limit of the fission product barriers as described in the FSAR being exceeded or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

No new methods of evaluation were used in the temporary relocation of mechanical, electrical, and structural interferences in support of the Unit 1 SG/RVCH replacement outage or in the permanent removal of two sections of the façade and relocation of the Reactor Building Purge exhaust duct on the exterior of the Reactor Building.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-028

I. OVERVIEW / SIGNATURES

Facility: **ANO Unit 1**

Document Reviewed: ER ANO 2005 0481 000 NCP, ERCN6 **Change/Rev.:** 0

System Designator(s)/Description: SW

Description of Proposed Activity:

Permanent removal of the tarpaulin atop the articulated slabs on the ECP spillway.

Temporary installation of components to reduce ECP level.

Check the applicable review(s): **(Only the sections indicated must be included in the Review.)**

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-028)**	**Sections I, II, and IV required**

Preparer: Dan H. Williams / **ORIGINAL SIGNED BY DAN WILLIAMS** / EOI / SA / 7-12-05
Name (print) / Signature / Company / Department / Date

Reviewer: Dan P. Hale / **ORIGINAL SIGNED BY JAMES HALE** / ENS-EOI / MCS / 7-12-05
Name (print) / Signature / Company / Department / Date

OSRC: J.N. Miller / **ORIGINAL SIGNED BY J.N. MILLER** / 7-14-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See LI-101 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	ANO-1 §1.7.3, ANO-1 §9.3.2.4, ANO-2 §9.2.5.2.1.1
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the LBD change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113.			

[1] If "YES," see LI-101. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 evaluation.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ **Yes** ☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113, if applicable. If obtaining NRC approval, document the change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.

3. **Basis**

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

None of the Operating License documents listed have a level of detail that includes the type of information involved in this ERCN.

The equipment proposed to lower the water level would change ANO-1 SAR Fig. 9-33 and ANO-2 SAR Fig. 9.2-11 if it was permanent. These two Figures correspond to Design Dwg. C-69 which also shows the tarpaulin ("impervious membrane fabric"). However, the current SAR figure version does not show the tarp.

The existence of the tarpaulin ("impervious membrane fabric") is explicitly mentioned in ANO-1 SAR sections 1.7.3 and 9.3.2 and ANO-2 SAR section 9.2.5.2.1.1, in all three places indicating that it has been placed above the articulated slabs to "deter erosion".

4. **References**

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

Electronic search method used: Keywords:

None

LBDs reviewed manually:

ANO-1 SAR

ANO-2 SAR

5. **Is the validity of this Review dependent on any other change?** ☐ **Yes** ☒ **No**

If "YES," list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any new or different chemicals that are currently not authorized for use by the state regulatory agency?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☒	☐	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117 for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered “yes,” a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility’s security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility’s telephone or security radio systems?

The Security Department answers the following question if one of questions C.1 through C.10 above was answered “yes.”

Is a change to the Security Plan required? ☐ **Yes** ☐ **No**

Attach to this 50.59 Review or reference below documentation for accepting a “yes” answer for any of Questions C.1 through C.10, above.

Name of Security Plan reviewer (print / Signature / Data

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING
(NOTE: This section is not applicable to Grand Gulf or Waterford 3 and may be removed from 50.59 Reviews performed for Waterford 3 proposed activities.)

If any of the following questions is answered "YES," a 72.48 Review must be performed in accordance with NMM Procedure LI-112 and attached to this 50.59 Review.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the ISFSI including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☒	☐	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power that could potentially impact cask loading or storage activities?
9.	☐	☒	Involve a change to the Fuel Building ventilation that could potentially impact cask loading or storage activities?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Involve fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components power that could potentially impact cask loading or storage activities?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

IV. 50.59 EVALUATION

<u>License Amendment Determination</u>

Does the proposed Change being evaluated represent a change to a method of evaluation <u>ONLY</u>? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.	☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?	☐ Yes ☒ No

BASIS:

The ECP is only for accident mitigation and cannot cause an accident.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?	☐ Yes ☒ No

BASIS:

The purpose of the tarpaulin (impervious membrane fabric) as described in the SAR is to deter erosion. Experience has shown that the tarp is not effective at eliminating erosion of the underlying clay embankment due to the amount of water that infiltrates under the tarp and flows through crevices that naturally exist between each of the articulated blocks. Additionally, the presence of the tarp has hindered inspections of the spillway, and thus has a net negative effect on maintaining the spillway such that degradation, when it occurs, can be quickly detected and repaired as necessary. Therefore, removal of the tarpaulin will decrease the likelihood off occurrence of a malfunction of the ECP.

The following precautions are being taken to prevent reduction of ECP level below that which is needed to support the analytical justification for the 30 day mission function of the ECP. These precautions assure that there is no more than a minimal increase in the likelihood of occurrence of a malfunction of the ECP while the level reduction equipment is in place.

- Prior to placing either a pump or siphon suction hose in the ECP, both Shift Managers shall be notified.
- Chemistry has been contacted and did not identify any concerns or limitation on discharging ECP water to Lake Dardanelle, but did request verbal notification (contact: Mike Prock) prior to initiation of the water discharge. Chemistry also requested that if possible, the suction source should be approximately 2 feet or more below the pond surface. Notify Chemistry as requested.
- The preferred method is to use an 8” or smaller hose to siphon water from the ECP to the stilling basin. A rope should be tied to each end of the siphon hose. The pond side rope may be used to break suction immediately if called for by Ops or because ECP has lowered sufficiently to facilitate required work on the spillway. The stilling basin side rope may be used to adjust the flow rate by raising the elevation of the hose discharge.
- An acceptable alternative is to use not more than two @ 500 gpm or less portable pumps to take suction from the pond and move it to the stilling basin.
- After validating the rate at which the ECP is dropping, both the pumps and the siphon may be used if both on duty Ops Shift Managers concur with a request to increase the flow rate.
- The siphon and/or pumps must be continuously attended by a dedicated person, in contact with the Control Room via cell phone or radio. The dedicated watch shall be briefed on the safety and regulatory significance of an unplanned LCO entry which could occur levels were dropped below required limits. Contact shall be verified every 15 minutes for one hour following initiation of draining and following any change in flow rate. (Radio or cell phone acceptable.) After one hour, contact shall be verified at a rate to be determined by Ops, but not less than once per hour. If contact is lost due to communication difficulty then siphoning and pumping shall be discontinued until contact can be reestablished.
- Discontinue siphoning and/or pumping at or before reaching an indicated level of 5.5 feet, or immediately if directed by Ops for any reason.
- In addition to the above dedicated watch, the work evolution shall adhere to any additional monitoring requirements requested by Ops personnel, including observation by or assistance from trained operators if requested.
- The estimated flow rate for an 8” siphon hose is somewhere between 500 and 2000 gpm. Very conservatively assuming 6000 gpm for the hose plus the two pumps (more than twice the estimated maximum rate), it would take over 1 hour to lower the ECP about 1”. Using the siphon hose alone, a realistic estimate is that it will take something between 3 hours and 12 hours to lower the ECP by 1”. (Note, the siphon rate may be easily reduced by raising the elevation of the hose discharge.) Using just the two pumps, it will take over 6 hours to lower the ECP.
- Upon initiating or increasing flow rates, the rate achieved shall be estimated and communicated to Operations shift managers (both units).
- Once the ECP desired level reduction is achieved, the pumps and/or siphon shall be moved to the stilling basin area, and Ops shall be notified that ECP level is no longer being reduced by the above described means.
- If area rains raise the ECP level, the above steps may be repeated, but only with concurrence from Ops and careful coordination with Ops as described above.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

Because, as discussed in the response to question 2, there is no effect on the capability of the ECP to perform its 30 day mission function, the consequences of any accident in whose mitigation the ECP is anticipated to play a role will not increase above those currently evaluated in the SAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The only malfunction mitigated by the ECP is a malfunction of ANO access to the water inventory normally in Dardanelle Reservoir. Because, as discussed in the response to question 2, there is no effect on the capability of the ECP to perform its 30 day mission function, the consequences of a malfunction of ANO access to the water inventory normally in Dardanelle Reservoir will not increase above those currently evaluated in the SAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The ECP is only for accident mitigation and cannot cause an accident.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

As discussed in the response to question 2, there is no effect on the capability of the ECP to perform its 30 day mission function. However, the results of a failure of the ECP as a single failure (no accompanying failure of access to Dardanelle Reservoir) are inherently evaluated in ANO-1 SAR table 9-15 and ANO-2 SAR 9.2-5. The physical remoteness of the ECP spillway from other (other than the ECP) structures, systems or components important to safety and the lack of any functional or support interface between other (other than the ECP) structures, systems or components important to safety, and the tarpaulin to be removed and the level reduction equipment, prevent the possibility of this change creating a possibility for a malfunction of a structure, system or component important to safety.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

The ECP can play a direct role in maintaining design basis limits for fuel cladding and reactor building/containment through its supply to the Service Water system and subsequent function as heat sink in the reactor building/containment coolers and decay heat coolers/shutdown cooling heat exchangers as well as an indirect role through component cooling of Emergency Diesel Generators, ECCS pumps and room coolers in which vital equipment is located. However, as discussed in the response to question 2, there is no effect on the capability of the ECP to perform its 30 day mission function. Therefore, no design basis limits for fission product barriers will be exceeded or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

This change does not involve any methods of evaluation.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-029

I. OVERVIEW / SIGNATURES

Facility: ANO Unit 1

Document Reviewed: CSTD Entry for FIC-1207 [RCP Seal Injection Control Valve] Controller
Change/Rev.: N/A

System Designator(s)/Description: Seal Injection

Description of Proposed Activity:

Conduct of Operations, 1015.001 Attachment A.5, Component Status Tracking Database (CSTD) Quarterly Review requires a condition report be written to request a 10 CFR 50.59 review be performed for components in the Unit 1 CSTD that have been out of position for greater then nine months. This review was performed to meet that requirement because FIC-1207 Controller was placed in manual due to Seal Injection Flow oscillations. The component was placed in the CSTD on 8/3/2005.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-029)**	**Sections I, II, and IV required**

Preparer: James Crabill, PE / **ORIGINAL SIGNED BY JAMES CRABILL** / EOI / OPS-1 / 8-24-05
Name (print) / Signature / Company / Department / Date

Reviewer: Steve Bennett / **ORIGINAL SIGNED BY STEVE BENNETT** / EOI / Licensing / 8-25-05
Name (print) / Signature / Company / Department / Date

OSRC: J.N. Miller / **ORIGINAL SIGNED BY J.N. MILLER** / 9-1-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See LI-101 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	Section 9.1.2.1
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the LBD change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113.			

[1] If "YES," see LI-101. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 evaluation.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR? ☐ **Yes** ☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113, if applicable. If obtaining NRC approval, document the change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.

3. Basis

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

System Function and Operation

Seal injection flow to the reactor coolant pumps is established and maintained utilizing an air operated control valve, CV-1207. CV-1207 is positioned by FIC-1207 using a signal derived from demanded flow (in automatic) or a manual valve position demand (in manual). A toggle switch is provided on the operator control station for manual valve positioning. Operating procedure OP-1104.002 allows the use of automatic or manual valve operation as desired by Operations. Total RCP seal injection flow is set at 32 – 40 gpm. Automatic operation is selected if desired. The valve is currently not operated in Automatic due to flow oscillations caused by mechanical binding in the valve. When in Automatic, valve operation is sticky and sluggish which causes valve overshoot. RCP seal injection is being maintained in manual at approximately 37 gpm.

RCP seal injection and return are Non-Safety Functions of the Make-up and Purification/High Pressure Injection System.

Operating License/Technical Specifications

The placement of the FIC-1207 controller in manual due to flow oscillations is below the level of detail in the Operating License and Technical Specifications. The current configuration of FIC-1207 controller does not make any statements in the Operating License or Technical Specifications untrue.

FSAR

The software word search and manual review identified a FSAR impact. Section 9.1.2.1 states that seal injection flow is automatically controlled to the desired rate.

Test or Experiment

The current position of the FIC-1207 controller in manual only changes the control mode of the controller. This is an allowable mode per the normal operating procedure, 1104.002. This change in control mode does not constitute a test or experiment not described in the FSAR.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

Electronic search method used:	Keywords:
50.59 Unit 1	Seal injection flow, CV-1207, FIC-1207

LBDs reviewed manually:

ANO-1 SAR Section 9.1

5. Is the validity of this Review dependent on any other change? ☐ **Yes** ☒ **No**

If "YES," list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any new or different chemicals that are currently not authorized for use by the state regulatory agency?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117 for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following question if one of questions C.1 through C.10 above was answered "yes."

Is a change to the Security Plan required? ☐ **Yes** ☐ **No**

Attach to this 50.59 Review or reference below documentation for accepting a "yes" answer for any of Questions C.1 through C.10, above.

Name of Security Plan reviewer (print / Signature / Data

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

(NOTE: This section is not applicable to Grand Gulf or Waterford 3 and may be removed from 50.59 Reviews performed for Waterford 3 proposed activities.)

If any of the following questions is answered "YES," a 72.48 Review must be performed in accordance with NMM Procedure LI-112 and attached to this 50.59 Review.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the ISFSI including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power that could potentially impact cask loading or storage activities?
9.	☐	☒	Involve a change to the Fuel Building ventilation that could potentially impact cask loading or storage activities?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Involve fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components power that could potentially impact cask loading or storage activities?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   Analyzed accidents in the FSAR Chapter 14 were reviewed. No accident was identified in the FSAR that is due to or related to the FIC-1207 controller placed in the manual mode of operation. Overall seal injection system performance is enhanced with the flow controller in manual due in order to achieve a more stable seal injection flow.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The ability to control the seal injection flow rate with FIC-1207 controller placed in the manual mode of operation is an original design configuration option and is allowed by the normal operating procedure. A more stable operation of the seal injection system will ensure continued performance and reliability of the reactor coolant pump (RCPs) seals. The mode of operation of the controller has no significant impact on the operation of the Makeup/High Pressure Injection pumps supplying seal injection due to the small amount of flow provided to the seals in conjunction with the relatively large makeup flow and available pump recirculation capability.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   As stated above, the analyzed accidents in the FSAR Chapter 14 were reviewed. No accident was identified in the FSAR where the radiation dose consequences altered as a result of or related to FIC-1207 controller being placed in the manual mode of operation. The change in mode of operation of FIC-1207 from Auto to Manual has no impact on any equipment important to safety. RCP seal injection and return are Non-Safety Functions of the Make-up and Purification/High Pressure Injection System.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The change in mode of operation of FIC-1207 from Auto to Manual has no impact on any equipment important to safety. RCP seal injection and return are Non-Safety Functions of the Make-up and Purification/High Pressure Injection System. Considering this, there is no increase in radiological dose consequences.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The change in mode of operation of FIC-1207 from Auto to Manual has no impact on any equipment important to safety and is not an accident initiator. RCP seal injection and return are Non-Safety Functions of the Make-up and Purification/High Pressure Injection System. There is no change to the function of the system with the manual mode of operation of FIC-1207. This configuration is allowed by the normal operating procedure.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The change in mode of operation of FIC-1207 from Auto to Manual has no impact on any equipment important to safety. RCP seal injection and return are Non-Safety Functions of the Make-up and Purification/High Pressure Injection System. The normal operating bands for seal injection flow are maintained in the manual mode of operation and therefore the overall operation of the system and potential failure effects are unaffected.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?

☐ Yes
☒ No

BASIS:

RCP seal injection and return are Non-Safety Functions of the Make-up and Purification/High Pressure Injection System.

The change in mode of operation of FIC-1207 from Auto to Manual does not affect the fuel cladding parameters, Reactor Coolant System boundary parameters or limits, and has no affect on the containment design pressure.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☒ No

BASIS:

This mode of operation of FIC-1207 from Auto to Manual does not change any calculations or methodologies. As determined by the screening, the mode of operation of FIC-1207 does not have any impact on any method of evaluation as described in the FSAR.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-030

I. OVERVIEW / SIGNATURES

Facility: ANO - Unit 1

Document Reviewed: ER-ANO-2002-1381-000 ANO1 ROTSG Design/ Qualification **Change/Rev.:** 0

System Designator(s)/Description: RCS/(Reactor Coolant System) (E24A and E24B)

Description of Proposed Change:

1. Background

In support of the Arkansas Nuclear One – Unit 1 (ANO-1) Steam Generator / Reactor Vessel Closure Head (SG/RVCH) Project, this ER documents the design and qualification of the AREVA enhanced once through steam generators (EOTSGs) and assesses potential effects on plant systems including the Nuclear Steam Supply System (NSSS). The original steam generators installed in ANO-1 were once through steam generators (OTSGs) manufactured by Babcock & Wilcox. The enhanced once through steam generators are manufactured by Framatome ANP, an AREVA and Siemens Company. The AREVA terminology of *enhanced* once through steam generators (**E**OTSG) is used interchangeably with the term *replacement* once through steam generators (**R**OTSGs).

The ROTSG occupies essentially the same physical envelope as the OTSG. Differences between the OTSG and ROTSG are intended to improve the operation, maintainability, and accident performance. Design differences are discussed below. There are no significant changes to the physical interfaces with the reactor coolant, main steam, feedwater, or other connected systems. AREVA performed, verified and approved the analyses and design in accordance with their quality assurance (QA) program. AREVA performed the evaluations and analyses for the NSSS and other connected systems with the ROTSGs to demonstrate the ROTSGs will support operation of ANO-1 with no adverse safety impact. As discussed in Section II.B, a License Amendment Request (LAR) was submitted by Entergy on September 15, 2004 and was approved by the NRC on August 10, 2005. This LAR was primarily the culmination of incorporation of the NEI 97-06 program to update the steam generator tube surveillance requirements, but also included adjustments to address OTSG replacement.

The Arkansas Nuclear One Unit 1 (ANO-1) RCS is a two loop system with each loop consisting of two reactor coolant pumps (RCPs) and one steam generator. The main function of the RCS is to remove the heat from the nuclear fuel and transport that heat to the steam generator for use in steam production. Reactor coolant flows past the nuclear fuel and through the tubes of the steam generator. Feedwater from the Main Feedwater (FW) System is supplied to the shell side of the steam generator. As the reactor coolant travels down through the steam generator tubes, heat is transferred to the feedwater which is converted to superheated steam. In addition to serving as a heat transport medium, the coolant also serves as a neutron moderator, reflector, and a solvent for soluble neutron poison (boron in the form of boric acid). RCS pressure is controlled by the use of electric heaters and a spray line in the pressurizer. Coolant flow from the core is directed to the top of the steam generators via a 36 inch inner diameter pipe (hot leg). As the coolant exits the steam generator, it is split into two separate 28 inch lines (cold legs). Each cold leg directs the coolant to the suction of the RCPs which pump the coolant back through the core where the process starts again.

The components of interest in ER-ANO-2002-1381-000 are the once-through steam generators. The once-through steam generator supplies superheated steam and provides a barrier to prevent fission products and activated corrosion products from entering the steam system. The steam generator is a vertical, straight tube, tube and shell heat exchanger which produces superheated steam at an approximately constant pressure over the power range. Reactor coolant flows downward through the tubes and transfers heat to generate steam on the shell side. The high pressure (reactor coolant pressure) parts of the unit are the upper and lower heads, the tube sheets, and the tubes between the tube sheets. Tube support plates maintain the tubes in a uniform pattern along their length. The unit is supported by a skirt attached to the bottom head. The shell, the outside of the tubes, and the tube sheets form the boundaries of the steam producing section of the vessel. Connected to a MFW distribution ring external to the shell, there are 32 feedwater nozzles which supply feedwater around the periphery of the shell. Within the shell, the tube bundle is surrounded by 2 cylindrical shrouds. There is an opening between the shrouds just above the centerline of the feedwater inlet nozzle elevation to provide a path for steam to afford contact heating as feedwater enters the lower shroud region which is the feedwater inlet heating zone. The annulus formed by the upper shroud and the shell routes the superheated steam down to two MS outlets near the bottom of the upper shroud. This steam annulus provides heating of the upper portion of the shell to minimize tube to shell differential thermal expansion during normal operations.

If required, emergency feedwater is supplied to the steam generator through 7 emergency feedwater nozzles connect to a distribution ring located at the top of the steam generator, and can be used to assure natural circulation of the reactor coolant following the unlikely event of the loss of all RCPs. The ROTSG design restores the 7th nozzle that was removed from service in the OTSGs because of concerns with flow in the open tube lane, which is eliminated in the ROTSG.

Retaining rings in the lower plenum of both steam generators in the area of the cold leg outlet nozzles allow the use of nozzle dams during refueling operations. Nozzle dams are used to prevent water from entering the steam generator lower head when the primary piping and refueling canal are full of water, thus allowing concurrent inspection and maintenance of steam generators while other refueling and reactor maintenance operations are occurring.

The change to the plant being reviewed in this evaluation is the replacement of the original ANO-1 Babcock & Wilcox supplied OTSGs with replacement ROTSGs designed and fabricated by Framatome-ANP. Due to various corrosion mechanisms, the existing OTSGs are experiencing degraded performance. The replacement ROTSGs are fabricated of advanced materials which have superior corrosion resistance as compared to the OTSGs. ER-ANO-2002-1381-000 does not involve field work.

Engineering Request (ER) responses ER-ANO-2002-1381-000 through ER-ANO-2002-1381-013 encompass the design and qualification of the ROTSGs as follows:

ER Number	ER Title
ER-ANO-2002-1381-003	ANO-1 RCS Impact Review
ER-ANO-2002-1381-004	ANO-1 Electrical / I&C Systems and Components Impact Review
ER-ANO-2002-1381-005	ANO-1 NSSS and Components Impact Review
ER-ANO-2002-1381-006	ANO-1 Documentation Impact Review due to change in EOTSG ASME Secondary ASME Code
ER-ANO-2002-1381-007	ANO-1 Civil/Structural Components Impact Review
ER-ANO-2002-1381-008	ANO-1 Piping Systems and Components Impact Review
ER-ANO-2002-1381-009	ANO-1 ROTSG EOTSG Unaffected Systems ER
ER-ANO-2002-1381-013	ANO-1 Umbrella ER – Systems Review Process

Related ER responses have been prepared to perform the removal of the OTSGs, installation of the ROTSGs, and the restoration of all affected plant structures, systems, and components (SSCs) as follows:

ER Number	ER Title
ER-ANO-2002-1078-012	Main Steam Piping and Supports
ER-ANO-2002-1078-013	Main FW and EFW Piping and Supports
ER-ANO-2002-1078-014	2” and Under Piping and Supports
ER-ANO-2002-1078-015	OTSG Removal & ROTSG Installation
ER-ANO-2002-1078-016	Insulation
ER-ANO-2002-1078-018	Interference Removal and Replacement
ER-ANO-2002-1078-020	Tubing/Supports and Instrumentation

This 10 CFR 50.59 evaluation applies to ER-ANO-2002-1381-000, ANO-1 ROTSG Design / Qualification which contains several interdisciplinary ER responses (ER-ANO-2002-1381-003, -004, -005, -006, -007, -008, -009, and -013). Separate 10CFR50.59 evaluations have been prepared for the installation ER responses associated with the ANO-1 SG replacement.

ER-ANO-2002-1381-000 is a compilation of various daughter ER responses noted above which provide the basis for the acceptability of the ROTSG design for operation. These ER responses evaluated the ROTSGs as components as well as evaluated the operation of plant systems which could be affected by any changes in plant operating parameters due to the ROTSGs.

The ROTSGs are designed to facilitate any future plant power uprate and thus, are capable of being operated at an uprated power of 3031 MWt (**OPEN ITEM FIV analyses for PUR is not complete – ER Hold Point 10.1**). However, this 10 CFR 50.59 review does not address any power uprate to ANO-1 and assumes the unit is operating at the 1R19 licensed power of 2568 MWt. A separate 10 CFR 50.59 will be prepared to address power uprate issues at a later date as necessary.

The ER responses ER-ANO-2002-1381-000 demonstrates that the ROTSGs and all affected plant systems will continue to meet current design and licensing requirements. System testing following installation of the ROTSGs will ensure that all SSCs will support Cycle 20 operation. All post-installation test selection, procedures, and acceptance criteria are outside the scope of this 10 CFR 50.59 evaluation.

2. OTSG/ROTSG Design Differences

The ROTSGs have been specifically designed to minimize the impact of the replacement SGs upon current plant systems and operations to the extent possible while making necessary design improvements. The ROTSGs are designed to retain the major design functions of the OTSGs, namely:

- Transfer heat generated by the primary system to the secondary system to support electrical power generation under forced flow conditions.
- Provide superheated steam at design flowrates to meet secondary plant requirements for electrical power generation.
- Provide the necessary heat transfer for the RCS during anticipated operational occurrences and analyzed transients.
- Provide the necessary heat transfer for core decay heat removal and plant cooldown during forced flow (reactor coolant pumps available) conditions or under natural circulation conditions.
- Provide a primary coolant boundary to prevent the release of fission products to the environment.
- The SG secondary side components (MFW header assembly, EFW header assembly, tubesheets, shell, nozzles, tubes) serve as a containment boundary (closed system inside containment).

However, differences between the two SG types have the potential for licensing basis impact. These differences and their disposition are as follows:

- The original code of record for the ANO-1 OTSGs was ASME Section III, 1965 Edition with summer 1967 Addenda [Ref. 21]. The ROTSGs are designed to the ASME 1989 Edition no Addenda. The material properties for the Alloy 690 material (refer to table below) were taken from ASME Section III, 1998 Code through 2000 Addenda [Ref.15]. This has been appropriately reconciled in accordance with ASME Code Section XI with no adverse impact identified [Ref. 7]. Structural and seismic analyses of the ROTSG demonstrate that the primary and secondary side pressure boundary components satisfy ASME III, Division 1, Class 1 and Class 2 design requirements [Ref. 15].
- The replacement main and emergency feedwater risers, elbows, and nozzles are designed and fabricated using Alloy 690 in accordance with ASME Code Case N-725, which describes the use of the nickel alloy in Class 2 applications. Use of this Code Case for the MFW and EFW piping is acceptable in accordance with 10 CFR 50.55a (d)(1).
- The secondary side design pressure and temperature of the ROTSGs are 1150 psig and 605°F, respectively [Ref. 5]. These are greater than the OTSGs values of 1050 psig and 600°F, respectively [Ref. 21]. However, the design pressures and temperatures of the interfacing systems (MFW, EFW, MS, and all vents and drains) remain unchanged. Additionally, the ROTSGs operate within the original SG design limits (e.g. pressures, temperatures, steam quality, primary system flow, feedwater flow). All overpressure protection devices and setpoints (i.e., main steam safety valves) remain unaffected and bound ROTSG operation [Ref. 22].
- Although the ROTSG was designed to specifically mimic the OTSG, the ROTSG has a slightly smaller primary side volume. According to Table 4-4 of the ANO-1 FSAR, the hot primary side volume of the OTSG is 2030 ft^3 per generator. The hot primary side volume of the ROTSG has been calculated to be 2026 ft^3, a primary volume decrease (SGs only) of 0.2% [Ref. 23]. This results in a total RCS volume decrease of approximately 0.07%. The ROTSG impact upon the post accident sump level, post accident pH, and NaOH requirements has been determined to be insignificant due to the small nature of this volume change [Refs. 9, 10]. There is no significant effect of this volume change on the accident analyses reviewed in Chapter 14 of the FSAR [Ref. 11].

- The ROTSGs have a larger downcomer which results in a greater secondary volume than the OTSGs (3666 ft^3 ROTSG cold volume compared to the OTSG cold volume of 3264 ft^3) and a corresponding larger secondary inventory at the various power levels. The ROTSG orifice plate flow area is adjusted to provide ROTSG normal and startup level indications which closely match the OTSG level indications that were present when the OTSGs were new and unplugged [Ref. 12]. Additionally, the effect of this larger secondary inventory on accident analyses has been evaluated [Ref. 11]. Plant SSCs remain within their design bases with the ROTSGs.
- The shell of the ROTSG is fabricated from forged SA-508 Cl. 3A cylinders. This material has a higher strength than the OTSG shell material, allowing for reduced thickness of the shell. Stress analyses have verified the acceptability of this reduced shell thickness [Ref. 15].
- The ROTSGs have a smaller metal mass than the OTSGs which results in the dry weight, flooded weight, and operating weight of the ROTSGs being less than the OTSGs. Correspondingly, the dry, flooded, and operating SG center of gravity is lower for the replacement SGs [Ref. 24]. The change in shell material and thickness reduces the total 100% power operating weight of the SGs from approximately 637 to 578 tons and the dry weight of the SGs from 570 to 502.3 tons. The effects of this weight and center of gravity change on the RCS piping and supports and on the surrounding containment structures (cavity floor and walls, etc.) have been evaluated [Refs. 13, 14] and determined to be bounded by the existing analyses. This decrease in metal mass will have an effect upon the sensible heat stored in the RCS during normal operations. This sensible heat decrease is slightly offset due to the increase in secondary inventory of the ROTSGs. Due to the smaller overall thermal mass, the normal decay heat removal cooldown will maintain cooldown from decay heat initiation to 140°F within 14 hours after the installation of the ROTSGs. This change in thermal mass will have an insignificant effect upon the plant responses to normal trips and transients [Ref. 27].
- The increase in steam temperature due to the ROTSG installation could cause a slight increase in heat losses to the reactor building. The increase in losses from the steam lines was determined to be insignificant. The primary average temperature is unchanged (T_{ave} = 579°F). The physical size of the ROTSGs is very similar to the OTSGs and the insulation quality for the ROTSGs will be better than the OTSG insulation (refer to ER-ANO-2002-1078-016, Insulation for additional information). Therefore, the cooling requirements of the reactor building will not be changed due to the ROTSG installation [Ref. 25].
- The ROTSGs are fabricated with a flow restricting venturi integral with the steam outlet nozzles while the OTSGs have no venturis. The venturis require a longer steam outlet nozzle which results in the ROTSG steam outlet nozzle elbow having a bend radius of 27” as compared to the current OTSG steam outlet nozzle elbow bend radius of 36”. The shorter radius elbows are provided by ER-ANO-2002-1078-012, Main Steam Piping and Supports. The flow restrictor limits steam flow during the unlikely event of a steam line break accident. During a steam line break, steam flow will become choked at the flow restrictor rather than at the break location. This reduces piping loads, energy release rates, and internal component hydraulic loads. These venturis are fabricated of Alloy 690 material which reduces the potential for erosive corrosion mechanisms. During normal operation, the venturis do not restrict flow (the flowrate is significantly below the critical velocity) and cause very little pressure drop. In addition, the impact of the ROTSG replacement steam nozzle outlet elbows upon existing piping and support analyses has been evaluated and determined to be insignificant [Ref. 14].
- There are notable material differences between the two SG designs. Refer to the following table:

Comparison of Materials of Construction

Component	OTSG	ROTSG
Primary Pressure Boundary		
Primary Upper and Lower Heads	SA-533 Gr. B Cl 1	SA-508, Class 3a
Primary Upper and Lower Head and Nozzle Cladding	Stainless Steel	Stainless Steel Type 308L/Type 309L
Primary Inlet/Outlet Nozzles	A-508-64 Cl. 1	SA-508, Class 3a (integrally forged with heads)
Tubesheets	A-508-64 Cl. 2	SA-508, Class 3a
Tubesheet Cladding	Alloy 82 (Alloy 600 equivalent)	Alloy 52 (Alloy 690 equivalent)
Tubes	SB-163, Alloy 600	SB-163, Alloy 690

Secondary Pressure Boundary				
Cylindrical Shell		SA-516 Gr. 70		SA-508, Class 3a (forged)
Main and Emergency Feedwater Nozzles		SA-106 Gr. B		SB-564/SB-167, Alloy 690
Main and Emergency Feedwater Elbows and Risers		SA-106 Gr. B		SB-564/SB-167, Alloy 690
Main and Emergency Feedwater Header Rings		SA-106 Gr. B		SA-182 F22
Main Feedwater Spray Plate		SB-167, Alloy 600		SB-168, Alloy 690
Steam Outlet Nozzle		A-508-64 Cl. 1		SA-508, Class 3a
Steam Outlet Venturis		N/A		SB-168, Alloy 690
Internals and Supports				
Tube Support Plates		SA-515 Gr. 70		SA-240 Type 410
Tube Support Tie Rods		SA-306 Gr. 70		SA-479, Type 410
Primary Nozzle Dam Rings		SB-167, Alloy 600		SB-167, Alloy 690
Support Skirt		SA-533 Gr. B Cl. 1		SA-508 Cl. 3a

There are numerous material changes with the installation of the ROTSGs which make the replacement SGs less susceptible to various corrosion mechanisms. All new materials utilized in the replacement SGs are materials with proven industry records for use in pressurized water reactor (PWR) reactor coolant and secondary systems. Strength and toughness are the critical selection criteria for all pressure boundary materials. The materials comply with the ASME code Section II and Section III. The ROTSGs have tubes fabricated of Alloy 690, the industry standard for replacement steam generators. All primary side ferritic steel surfaces (primary side of the tubesheet and inside surfaces of the primary head) are clad with austenitic stainless steel (Type 308L/Type 309L) or Alloy 690 (Alloy 52) to prevent corrosion [Ref. 47]. The ROTSG EFW and MFW distribution system components carry feedwater at relatively high velocities and are designed to accommodate temperature gradients that occur between the feedwater, the steam, and the ROTSG shell. All portions of these components exposed to flow are fabricated from either Alloy 690 or 2¼ Cr-1 Mo low alloy steel, materials selected for their resistance to flow-assisted corrosion (FAC) [Ref. 48]. Materials for the various ROTSG internal components are carbon steel or a martensitic stainless steel. These materials are readily welded and have the required strength for their specific applications. Internal components in close proximity to tubing, such as tube support tie rods, are fabricated from a martensitic stainless steel material. Sufficient corrosion allowances are applied to each component to ensure that the components are compatible with all operating/shutdown conditions, including required chemical cleanings. All ROTSG components were demonstrated to maintain structural integrity under all service loadings (as required by ASME Section III) for the forty year design life [Ref. 15].

- The OTSG tubes were fabricated from Inconel 600 material while tubing material used in the ROTSGs is alloy 690 thermally treated (TT). The differences in thermal expansion and material strength of alloy 690 versus alloy 600 have been evaluated in the tube loading calculations and determined to have acceptable results [Ref. 15]. The alloy 690 exhibits high resistance to stress corrosion cracking in primary and secondary side environments. It has been utilized in most replacement steam generators in the United States. Evaluation of vibration and wear of alloy 690 tubing is evaluated in detail in section 3.
- The OTSG has carbon steel broached tube support plates. The ROTSG has stainless steel broached tube support plates that are thinner. The stainless steel support plates are less susceptible to the buildup of deposits between the tube support plates and tubes. Stress analyses have verified the acceptability of the structural adequacy of the thinner tube support plates [Ref. 15]. Evaluation of impacts of the change in TSP material on the vibration and wear of alloy 690 tubing is evaluated in detail in section 3.

- The OTSG has a partial row that was not tubed in order to allow for access to the interior of the tube bundle for secondary side tube inspection and sludge lancing. This open lane resulted in an area of low quality high density steam that has led to the degradation of tubes adjacent to the lane by moisture carried up though the lane where heat transfer is reduced, and the higher velocities in the lane reduce margins for flow-induced vibration (FIV) of the tubes adjacent to the tube free lane in the OTSG. Because of FIV concerns adjacent to the lane, the EFW nozzle located in line with the OTSG lane has removed from service. The tube free lane has been eliminated in the ROTSG by tubing all rows. This results in 66 additional tubes in the ROTSG and a slight increase in the primary side volume. This volume increase is offset by the flat bottom lower head. The net change in the RCS volume is a decrease of approximately 4 cubic feet. The ROTSG design also restores the 7th EFW nozzle as noted earlier. The additional tubes in the ROTSG essentially offset the lost heat transfer of the lower thermal conductivity of the alloy 690 compared to the OTSG, maintaining similar heat transfer coefficients. The tubed lane and design changes to the tube suport plate configuration at the periphery results in decreased by-pass flow, improving the amount of superheat and decreasing the potential for moisture to carry up to the upper tubesheet region. This latter effect decreases the potential for contaminates in the moisture from depositing on the tubes at the upper tubesheet and affecting the tubes.
- The main and emergency feedwater headers on the OTSG are carbon steel with flanged connections (although the MFW spray nozzzles were replaced with alloy 690 material due to erosion of the original carbon steel). The present headers present an erosion/corrosion concern. The flanged connections have had a history of leaking. The ROTSG feedwater headers are fabricated from an erosion/corrosion resistant material. The flanged joints are seal-welded to eliminate leakage problems. The ROTSG main and emergency feedwater headers retain the vertical riser configuration of the current systems that are not susceptible to water hammer. The EFW ring header connects to the ROTSG shell via 7 risers verses the 6 risers used by the OTSG (Ref. 17).
- The OTSG has a hemispherical shaped lower head. The ROTSG has a flat bottom lower head. This feature eliminates the need for a drain line at the bottom of the lower head. Deletion of the drain line eliminates a crud trap and the associated radiological/radiation dose concerns it presents.
- The ROTSG tube sheets are fabricated from SA-508 Cl. 3a material. The ROTSG tubes are flush welded to the primary face of the tube sheet and hydraulically expanded within the full thickness of the tubesheet to maximize mechanical strength and to minimize the tube-to-tubesheet crevice. The hydraulic expansion was designed to provide a room temperature preload that optimizes the normal operating and accident loadings on the tubes due to differential expansion between the tubes and shell. The tube to tubesheet weld is reduced in size from the OTSG weld which is external to the tubesheet. Stress analyses have verified the acceptability of this weld. [Ref. 15]
- The feedwater downcomer orifice arrangement in the ROTSG is slightly different from the arrangement in the OTSG to enhance the ability to adjust the orifice plate if required and provide finer adjustment capability. Stress analyses have verified the acceptability of this orifice arrangement. Thermal hydraulic modeling was performed to provide response for pressure drop and level indication similar to the original OTSGs. [Ref. 15]
- The lower shroud is supported differently in the ROTSG to maximize the flow area into the tube bundle. Stress analyses have verified the acceptability of this shroud arrangement [Ref. 15]. The ROTSG has additional tie rods between the tube support plates that are a different material and size than the OTSG tie rods. Stress analyses have verified the acceptability of this tie rod arrangement [Ref. 15]. Evaluation of impacts of the change in tie rod material on the vibration and wear of alloy 690 tubing is evaluated in detail in section 3.
- The tube to tubesheet weld is reduced in size from the OTSG weld which is external to the tubesheet. Stress analyses have verified the acceptability of this weld. [Ref. 15] The ROTSG has 7 EFW nozzles versus 6 on the OTSG. The effect of this change on the EFW flowrate was determined to be negligible.
- The ROTSG has 7 EFW nozzles versus 6 on the OTSG. The effect of this change on the EFW flowrate was evaluated in Ref. 17.
- The cold leg piping on the OTSGs will be cut just outside of the support skirt to facilitate removal of the OTSGs. The ROTSGs were manfactured with an additional section of cold leg piping attached. The original cold leg elbow was fabricated from formed and welded SA-516 Gr. 70 plate. The replacement elbows are a one piece SA-508 Cl. 3a forging. The acceptability of the cold leg elbow material was addressed in Reference 14.
- The ROTSG vents, drains and instrument connections are located in essentially identical locations with the exception that the ROTSG has eliminated the bottom head primary drain and some instrumentation taps are rotated a maximum of 11° azimuthally.

- The ROTSG has approximately the same configuration of manways and handholes, with the exception that the ROTSG will have fifteen 3" diameter tube support plate inspection ports and four (two 8 in. ID and two 3 in. ID) orifice plate adjustment locking handhloles and inspection ports. Handholes and inspection ports that are not routinely used have seal-welded diaphrams to prevent leakage. Primary manway sizes were increased from 16" to 18" to improve access for ISI.

3. Potential for New Failure Modes

As previously noted, the ROTSGs have been specifically designed to minimize the impact of the replacement SGs upon current plant systems and operations. It is recognized that the design improvements and material changes incorporated in the ROTSGs have the potential for introducing new failure modes. A number of failure modes were identified and addressed in Reference 5 and are summarized below.

Potential for New Tube Impact / Wear Failure Mechanisms – **ER HOLD POINT 10.1 – Pending review against revised FIV Report. Information will be revised as necessary through ERCN.**

The ROTSG tube support plates are positioned within the steam generator shroud at elevations the same as the OTSG which have shown to be effective to mitigate flow-induced vibration (FIV) and wear. Changes in the secondary internal configuration between the ROTSG and the OTSG (such as reduced flat land contact area and thinner tube support plates) have been evaluated for potential impact to result in an increase in wear and impact damage potential to the tubes due to FIV. Additionally, FIV analyses were performed to confirm that the tube bundle is adequately supported to avoid significant levels of tube vibration. Particular areas of emphasis are cross flow at the bottom and top spans of the tube bundle, vortex shedding resonance, and random turbulence excitation. A three-dimensional thermal hydraulic analysis was performed to derive a detailed flow distribution in the first, tenth, fifteenth and sixteenth spans of the ROTSG. The potential for fretting wear was assessed by non-linear FIV/wear analysis (Ref. 71 and 73). The FIV analysis was used to confirm that the tube bundle is adequately supported to prevent excessive tube wear due to FIV excitation mechanisms. Additionally, there are many design enhancements that improve the ROTSG FIV performance over the original OTSG. These include:

1. The shroud opening area at the steam exit region of the tube bundle was reduced. Therefore, flow loadings are reacted more at the tubesheet and are less distributed between the 15th TSP and tubesheet and there is less potential of exciting the tube for the flow induced vibration mechanisms.
2. The cross flow loadings in the first span were reduced by removing the water ports in the lower shroud entrance region, thereby distributing the feedwater flow across the entire periphery of the tube bundle.
3. The OTSG design had an open tube lane to allow inspection within the tube bundle. The open tube lane created flow velocities in the lane region that exceeded the general non-lane tube bundle flow velocities resulting in the OTSG tubes adjacent to the open tube lane experiencing the lowest margins for flow induced vibrations. The ROTSG design eliminated the open lane, thereby reducing the maximum flow velocities in the ROTSG and increasing FIV margins.
4. The ROTSG has stainless steel tube support plates, fabricated from material specification SA-240-TP410, that are less susceptible to corrosion product build-up and the resulting locked tube conditions.
5. The OTSG and the replacement ROTSG TSP thicknesses are 1.5 and 1.18 inch respectively. The thinner TSP thickness will provide lower flow losses, however the amount of available wear volume will be lower for the ROTSG from an analytical perspective. The vendor (FANP) has performed extensive testing on this variable and found that once the land length is greater than approximately 1.1 inch, then additional length offers insignificant improvements in the wear rate of the tube.

The results of all analyses demonstrated that all acceptance criteria were met, and that the tube bundle is adequately supported for the prevention of detrimental flow-induced vibration (Ref. 72). Additionally, it was demonstrated that the end of life wear of ROTSG tubes is acceptable. (Ref. 71)

Impact upon the tubes from loose parts can potentially damage the tubes, leading to tube failures. The existence of a loose part within a steam generator, whether originating from outside or within the unit, can cause significant damage to the steam generator, particularly to the tube bundle components. While this is also a concern with the OTSGs, the ROTSG design employs specific criteria to minimize the potential for loose parts. On the secondary side, most installed parts are weldments, others are securely captured. Where fasteners must be used, the design fastener material is specified with a chemistry that permits lock-welding of the component. Where internal fasteners that are not lock welded are used, the bolt or nut is locked in place with a corrosion resistant locking tab or the nut is designed to be locked by crimping it onto the corresponding stud. To avoid cracking of high strength bolting materials, fasteners with ultimate tensile strengths over 150 ksi are not used. While the impact resistance of Alloy 690 is similar to Alloy 600, the general probability of a loose part within the generator is reduced. The overall effect of these failure modes is a tube leak or tube rupture. As the primary and secondary operating conditions of ROTSG are essentially identical to the OTSG, the effect of these failure modes is essentially identical to the existing effects.

Of note is the recent operating experience of Duke Power's Oconee Nuclear Station Unit 1. ONS-1 has two once through steam generators designed and fabricated by Babcock and Wilcox International (BWI) which were put into service in 2003. The first inservice inspection of the steam generator tubing following this replacement started in April 2005. All of the inservice tubes were inspected full length with a bobbin coil. As a result of these inspections, approximately 11.5% of the tubes in steam generator A and 9.6% of the tubes in steam generator B had indications of wear at the tube support plate elevations. Most of the indications were located between the 9th and 11th tube supports in the periphery of the tube bundle. In addition, most of the indications are shallow (less than 20 percent through-wall) and all of the tubes had adequate structural and leakage integrity. The repair criteria implemented during the outage were 28% through-wall, resulting in 30 plugged tubes in steam generator A and 18 plugged tubes in steam generator B. All plugged tubes were stabilized for the full length of the tube. With respect to the causal mechanism, additional reviews of the original models and evaluations are being independently performed by Duke Energy. In addition, thermal hydraulic experts are looking at what could be causing the wear. A detailed design review is being performed to compare the old design of the steam generators to the new design. In addition, plant operating parameters are being reviewed to identify any differences from what was considered in the design of the replacement steam generators. AREVA and EOI are following these evaluations closely, and are evaluating the changes from the original OTSG design made by BWI against those made by AREVA to assess the potential commonality. These evaluations will be contained in the ER revision to be provided prior to heatup and included in a revised 50.59. The susceptibility of the ROTSGs to the flow conditions and characteristics causing classic tube wear has been evaluated as part of the design and analysis of the steam generators and found susceptible. As stated above, the maximum predicted end-of-life percent through wall degradation resulting from fretting wear and turbulence induced vibrations was determined to be acceptable for the design life of the ROTSGs (Ref. 71). New evaluations are on-going regarding the unexpected, non-classic wear occurring at Oconee. Additional information will be available this fall when Oconee 2 inspections of their ROTSGs will reveal if there is a unit specific or general aspect to the degradation mechanism.

Regardless of the conclusiveness that can be obtained regarding the Oconee 1 phenomenon, the ANO-1 Steam Generator Integrity Program, in accordance with NEI 97-06, invokes proven condition monitoring and in-service inspections that can identify this phenomena (e.g. leakage monitoring and ECT, respectively) and corrective actions can be taken (i.e., administrative limits requiring shutdown and tube plugging, respectively) prior to tube failure.

Potential for Tube Degradation Due to Tube-to-Shell Differential Temperature

Similar to the OTSGs, the ROTSG tubes are fabricated of nickel based alloy material while the shell of the generator is fabricated of carbon steel. The design is such that the tubing is locked into the tubesheets at each end of the steam generator and the shell sets the spacing of the tubesheets. Accordingly, when the steam generator experiences a heatup or cooldown transient, the differential thermal expansions between the tube and shell materials imparts loads onto the tubes. During fabrication, the steam generator tubes are installed under tension (OTSG:≈ 40 to 70 lbs; ROTSG: 100 – 325 lbf) to offset the compressive stresses developed during operation. The current OTSG guidance specifies a 60°F tube to shell ΔT (tubes hotter) and a 100°F tube to shell ΔT (tubes colder) normal limits as well as a 150°F tube to shell ΔT emergency limit (tubes colder). The thermal expansion of Alloy 690 (ROTSG tubing) relative to SA-508 Cl. 3a (ROTSG shell) is approximately 10% greater than that for Alloy 600 (OTSG tubing) relative to SA-516 Gr. 70 (OTSG shell). The inner radius of the ROTSG is greater than the OTSG as the shell of ROTSG is thinner. As the ROTSG is thinner than the OTSG, the ROTSG is not as stiff as the OTSG which offsets the increased expansion coefficient and in conjunction with the preload reduces the steady state tube loads compared to the OTSG. The thinner shell also allows the ROTSG shell to change temperature faster than the OTSG, resulting in reduced transient loads because at any given time in the transient, the ROTSG tube to shell ΔT is less than that of the OTSG as demonstrated in the tube analyses (Ref. 5). Therefore, the current tube to shell ΔT limits remain acceptable to prevent tube damage. These current differential temperature limits were demonstrated acceptable with a 10°F instrument uncertainty. In addition, the preload on the ROTSG tubes will preclude undesirable compressive loads at normal operational temperatures.

Potential for Different Material/Environment Combinations

The ROTSG contains new materials when compared to the OTSG. The use of new materials in the ROTSG construction could lead to a new degradation mechanism not experienced with the OTSG. The tubing material differences are discussed above. Also as discussed above, the tube support plates are fabricated of stainless steel (SA 240 Type 410) in the ROTSG as compared to the SA 515 Grade 70 carbon steel plates of the OTSGs. The use of stainless steel results in increased reliability of the plates due to the increased corrosion resistance, wear resistance, and strength. These Framatome-ANP Type 410 stainless steel broached tube support plates have been successfully operated for more than 20 years in recirc (U-tube) type SGs. As these plates are utilized in a low stress application, stress corrosion cracking of these plates is not a concern.

The OTSGs had carbon steel tube support plates (TSPs) in combination with carbon steel tubesheets. Thus differential thermal expansion was minimal. An area of concern for the ROTSG is the differential expansion of the 410 SS TSPs compared to the carbon steel tubesheets. The TSPs will expand less than the tubesheets which imposes a lateral load on the tubes and thus increases wear potential. This phenomenon has been evaluated in Ref. **(LATER-ER HOLD POINT 10.1)**

The clad material on the primary side of the ROTSG tubesheets is fabricated of type 52 wire (Alloy 690 equivalent) in contrast with the type 82 wire (Alloy 600 equivalent) of the OTSG. As discussed in Section 3.1.3.2 of Ref. 5, Alloy 690 material demonstrates a higher resistance to primary water stress corrosion cracking than the Alloy 600 material used in the OTSG and has a proven industry record of corrosion resistance. Therefore, no increased corrosion potential of this material/environment combination on the ROTSG is expected.

The ROTSG main and emergency feedwater distribution system components (feedwater nozzles, MFW diffuser plate, feedwater headers, risers and elbows) carry feedwater at high velocities and accommodate temperature gradients that occur between the feedwater, the ROTSG water, and the ROTSG shell. All portions of these components exposed to flow are fabricated from either Alloy 690 or CR -Mo. The comparable OTSG components are fabricated of carbon steel. The replacement of these carbon steel components with nickel alloy or CR –Mo components will significantly reduce the susceptibility of these components to flow accelerated corrosion and erosion. In addition, the welded configuration of the headers and risers will reduce the potential crevices in which corrosive buildup can occur.

Integral Flow Restricting Venturis in Main Steam Nozzle

The ROTSGs contain a flow restricting venturi integral to the main steam exit nozzle. The OTSGs do not contain any such device. These venturis are intended to restrict flow in the event of a main steam line break to allow for a longer dryout time for this event. For the effect of these devices upon the MSLB analysis, refer to ER-ANO-2002-1381-005 (Ref. 11). The venturis have no moving parts and are fabricated of Alloy 690 for erosion/corrosion resistance. During normal operation, the venturis do not restrict flow and cause very little pressure drop. No significant increase in vibration is expected due to the venturis during normal operation. Flow restrictors of this type are frequently used in this manner. (The ANO Unit 2 replacement steam generators contain flow restricting venturis at the steam outlet nozzle.) For licensing transient analyses (SAR Chapter 14 events), the venturis were not modeled in all transients but have been evaluated where there was a possibility that they would have an effect on the results of the analysis. No new failure modes or effects are anticipated due to the venturis.

Increased RCS Nickel Content Due to ROTSG Tube Passivation

The installation of the ROTSG will result in tubes which have not yet developed the strong passive oxide layer. In some plants who have replaced their steam generators, this resulted in increased nickel in solution and then plating out in crud on the fuel assemblies. During the initial plant startup following replacement, metal release rates, especially for nickel, will be higher than equilibrium release rates. This phenomenon will add to the cost burden by increasing personnel dose rate at the first refueling outage following replacement (1R20) due to the deposition of nickel on fuel rods and its subsequent activation to Co-58. However, once the oxide layer has developed on the ROTSG tubes, this phenomenon will diminish to the normal expected metal release rates. The use of zinc acetate at Southern Nuclear's Farley station confirmed that zinc is beneficial overall for radiation field control. More specifically, Farley 1 did not experience the radiation dose increase after steam generator replacement that other sites have experienced. An evaluation of a zinc injection program for radiation dose reduction at ANO-1 concluded that a low-concentration zinc injection program may be safely and effectively applied at ANO-1 (Refs. 19 and 45). A zinc injection system has been installed at ANO-1 and has been injecting low levels of zinc. Zinc injection is expected to be effective even if it has not been injected for the last few weeks of Cycle 19 and the first few weeks of Cycle 20 due to the slow buildup of nickel from the ROTSGs' tubes and the residual zinc remaining in the RCS (Ref. 80, 81). **ER HOLD POINT 10.5**.

Increase Fatigue Usage Factor Due to Lower Feedwater Temperature Limit

The allowable feedwater temperature limit for the steam generator MFW nozzle is being lowered to 135°F from 185°F as part of the steam generator replacement. The lower temperature feedwater could lead to increased fatigue cycles on the feedwater nozzle and a decreased time to fatigue failure of this joint. However, the ROTSG feedwater nozzles and thermal sleeves are fabricated of Alloy 690 (as opposed to the OTSG carbon steel) which has a superior fatigue resistance compared to carbon steel. In addition, the ROTSG Section III Code Certified Design Report has analyzed the nozzle for appropriate cycles at this temperature and verified acceptable fatigue usage factors for the 40 year design life of the steam generator (Ref. 70). Therefore, the reduction of the allowable feedwater temperature will not invalidate the design fatigue life of the ROTSG.

Increased Potential for Flow Induced Vibrations of Nozzles and Appendages

Operation with the ROTSGs results in slight changes in the secondary flow characteristics through the steam generators due to physical and flow parameter changes. These changes could increase the potential for flow induced vibrations of the instrumentation and EFW nozzles which protrude into the secondary flowpath. Five instrumentation nozzles are exposed to secondary-side flows and were evaluated for full power, steady state conditions (Ref. 74). The remaining instrumentation nozzles of the ROTSG do not protrude into the feedwater or into the steam annulus and therefore are not susceptible to flow-induced vibrations. The analyses showed that the instrumentation nozzles installed in the ROTSG steam and feedwater annuluses will have acceptable FIV performance and will not experience lock-in from vortex shedding during any steady state operating condition. Additionally, the instrumentation nozzles will not experience vortex lock-in during RCS power maneuvers or heatup/cooldown transients. An analysis was also performed to ensure that the instrumentation nozzles will not experience vortex lock-in during any of the elevated secondary-side flow conditions. The analysis demonstrated that the instrumentation nozzles located in the steam and feedwater annuluses have adequate structural integrity and will not experience vortex lock-in during normal/upset or faulted elevated-flow operating conditions. In addition, the EFW nozzles protrude through the secondary annulus and are exposed to secondary flow conditions. However, the physical size of these nozzles is such that the vortex shedding frequency will not approach the natural frequency of the nozzle. Therefore, the ROTSG EFW nozzles will not be subject to damaging FIV.

The shroud closure plug t-bolts are also located in the secondary flowpath. Analysis demonstrates that these subcomponents will have acceptable FIV performance and will not experience vortex lock-in for any steady state power levels including the uprated power level (3031 MWt) or during RCS power maneuvers or heatup/cooldown transients. Additionally, these subcomponents have adequate structural integrity for the normal/upset/faulted operating conditions. The stress loadings in the nozzle dam retaining ring and attachment weld are negligible. Therefore, there are no FIV concerns for these subcomponents (Ref. 74).

Additionally, it was demonstrated that the orifice plate is unlikely to lift due to the large hydraulic forces acting to hold the plate down (Ref. 75). The net force on the plate was consistently downward at both current and power uprate conditions with the orifice plate either fully open or closed. Therefore, the orifice plate is not likely to experience any significant flow vibrations during power operations.

Increased Potential for Missile Generation

No new unanalyzed types of missiles will be generated due to the ROTSGs. Possible missiles have been analyzed in the SAR for components located in the D-ring.

Combined Effects of Changes on Fatigue Evaluations

Combined effects of changes in the ROTSG design have resulted in a reduction in some of the design cycles for transients. These design changes have been reflected in changes to the Code analyses, SYE cycle tracking procedures, and SAR tables. **(ER HOLD POINT 10.3 - THESE ARE STILL CHANGING as a result of pressure/stress issue)**

4. Discipline Reviews

An expert panel of experienced engineers, knowledgeable in ANO-1 SSCs and plant operations, identified those systems which could be adversely affected by changes to system operating parameters due to operation with the ROTSGs [Ref. 63]. To assist in identifying these systems, the panel utilized a Parameter Impact List (Ref. 20) which provided a comparison of system and component parameters between the OTSGs and the ROTSGs. The systems listed below were selected for evaluation by the appropriate disciplines within the ER-ANO-2002-1381-000 daughter ER responses.

- Reactor Coolant System (RCS)
- Low Pressure Injection / Decay Heat Removal System (LPI)
- High Pressure Injection / Makeup System (HPI)
- Core Flood System (CF)
- Reactor Building Spray System (BS)
- Reactor Building Drains (RBD)
- Reactor Building Sump (RBS)
- Hydrogen Recombiners System (HR)
- Reactor Building Heating and Ventilation System (RBHV)
- Reactor Building (RB)
- Auxiliary Building (AB)
- Service Water System (SW)
- Chemical Addition System (CA)
- Sampling System (SS)
- Chilled Water System (AC)
- Emergency Feedwater System (EFW)
- Main Feedwater System (MFW)
- Plant Makeup System (PMU)
- Circulation Water System (CW)
- Condensate Storage and Transfer (CT)
- Condensate System (CS)
- Condensate Demineralizer System (DI)
- Nitrogen System (N2)
- Auxiliary Cooling Water (ACW)
- Main Steam System (MS)
- Reheat Steam System (RS)
- Gland Steam System (GS)
- Extraction Steam System (EX)
- Intermediate Cooling Water System (ICW)
- Fuel Oil System (FO)
- Feedwater Pump Lube Oil System (FPO)
- Integrated Control System (ICS)
- Emergency Feedwater Initiation and Control (EFIC)
- Reactor Protection System (RPS)
- Engineered Safeguards Actuation System (ESAS)
- Plant Computer System (EC)
- Diverse Reactor Overpressure System (DRPS)
- Diverse SCRAM System (DSS)
- Nuclear Instrumentation System (NI)
- Non-nuclear Instrumentation (NNI)
- Safety Parameter Display System (SPDS)
- Annunciator System (K)
- Smart Automatic Signal Selector (SASS)
- Vibration and Loose Parts Monitoring (VLPM)
- Isophase Bus Cooling (IBC)
- Isophase Bus System (IB)
- ATWS Mitigation System Actuation (ATWS)

- Turbine Generator System (TG)
- Emergency Diesel Generator System (EDG)
- 6.9 kV Switchgear System (H)
- Main, Unit, and Startup Transformers (XFMR)

Each of the systems above was evaluated for impact in separate calculations which feed into the respective ER responses listed in Section 1, above. ER-ANO-2002-1381-013 (Ref. 63) was created to serve as a roadmap for this task and describes the system evaluation process as well as how each of the system evaluations feed into discipline ER responses which, in turn, feed into the parent design and qualification ER response, ER-ANO-2002-1381-000.

The review for the design and qualification of the ROTSGs has been divided up into the applicable discipline responsibilities. ER-ANO-2002-1381-000 compiles input from all disciplines to become the complete design and qualification package for the operation and licensing of the ROTSGs. A summary of each discipline review is provided in the following sections:

Mechanical Discipline

The mechanical discipline review includes an extensive review of the ROTSGs as a component as well as the impact of the ROTSGs upon other plant SSCs. As described above, an expert panel of experienced engineers knowledgeable in ANO-1 SSCs and plant operations selected appropriate systems as those which could potentially be adversely affected by changes to system operating parameters due to operation with the ROTSGs. These systems were reviewed through various system impact reviews. A brief summary of the review results is discussed below.

Reactor Coolant System
There are no physical modifications to the RCS pressure boundary required covered by this ER which addresses operation and maintenance after installation. The operating pressure of the RCS will be the same as in previous cycles and well below the design pressure of the piping and ROTSGs. The location of the ROTSG primary nozzles is essentially identical to those on the OTSG, but the configuration is different as the nozzles are forged into the channelheads and an extension piece is welded to the forging to form the nozzle, vs. the OTSG which has the nozzles welded into the channelhead. The ROTSG cold legs also have the first RCS piping elbow welded to the extension piece for installation purposes. The full nozzle furnished by the component manufacturer will be field cut as a half elbow to accommodate installation. This configuration represents the current industry standard for nozzles. The nozzles and the primary pressure boundary of the ROTSG satisfy the structural requirements of ASME Section III, 1989 Edition with no Addenda, for all loading conditions [Ref. 15].

RCS component operation is impacted by the use of the ROTSGs. Detailed reviews were performed to assess the effects of the ROTSGs upon normal operations of the RCS components. Analyses were completed to verify acceptable RCS transient response with the ROTSGs installed [Ref. 27]. The significant issues related to the major RCS components are summarized below. Refer to ER-ANO-2002-1381-003 (Ref. 5) for additional details regarding the review of normal plant operation of the RCS with the ROTSGs.

Reactor Vessel
The 10 CFR 50 Appendix G reactor vessel nil-ductility temperature (NDT) and low-temperature over pressure (LTOP) limits have been evaluated for impact due to the ROTSGs. There are no changes to the reactor vessel NDT or LTOP limits due to the ROTSGs. The net positive suction head (NPSH) requirements and seal staging for the reactor coolant pumps (RCPs) are also not affected due to the ROTSGs and the current operational pressuretemperature window remains valid [Ref. 5].

The slight increase in flow from the ROTSGs will lead to a slightly better heat transfer coefficient and higher departure from nucleate boiling ratio (DNBR) within the core. Therefore, from a DNB safety margin perspective, the ROTSGs are conservative. This is evaluated in the Cycle 20 reload analyses. [Ref. 77, 78, 79]

Pressurizer
The installation of the ROTSGs does not affect the normal operation of the pressurizer, heaters, or spray. Normal operating pressure is unchanged and, as RCS T_{ave} is unchanged, pressurizer water levels are controlled at the same range as with the OTSGs. The pressurizer safety valves are credited with preventing the over-pressurization of the RCS. As the design pressure of the primary side of the ROTSGs are the same as the

OTSG (2500 psig), the current pressure setpoints for these valves is acceptable. There is no physical change to any of the piping or components associated with the pressurizer safety valves. The overpressure protection for the RCS with the ROTSGs has been evaluated and demonstrated acceptable for the limiting transient (startup event). The ASME overpressure protection report has been updated to verify acceptable operation with the ROTSGs [Ref. 22].

The slight decrease in primary system volume will have no appreciable effect upon the ability of the pressurizer to compensate for RCS fluid volume changes during system normal transients. The primary volume influences the shrink and swell of the RCS when primary to secondary heat transfer is unbalanced. This, in turn, affects the rate of pressure changes in the RCS as the RCS fluid expands into or contracts from the pressurizer. The hot primary volume change of the system is 0.07% when compared to the RCS volume with the OTSGs. This small volume change is inconsequential upon pressurizer operations and associated RCS pressure control.

Reactor Coolant Pumps/RCS Flow Rate
The installation of the ROTSGs will slightly increase the normal RCS flow rate above the current RCS flow rate with the current percentage of plugged OTSG tubes. The flow rate is increased by the ROTSG installation to a flow rate that is close to the flow rate that existed for the OTSGs in the unplugged condition. The change in fuel type that is planned to begin in Cycle 20 (outside of the scope of this evaluation) will cause a slight reduction in the flow rate, but calculations that were completed as a part of the Cycle 20 reload analysis that considered the effects of ROTSG installation *and* fuel type change that verified the flow rate will be acceptable. There is no change to the fourth pump startup temperature *due to the ROTSGs* [Ref. 5], however the change in fuel type will result in a change to the fourth pump startup temperature for Cycle 20. See the Cycle 20 reload analyses for the detailed evaluation of the effects of the change in fuel type. [Ref. 77, 78, and 79]

There will be no effect upon the RCP seals as a result of the installation of the ROTSGs. This is because the pump operating speed, RCS fluid operating temperatures and pressures are not significantly changed.

The reactor coolant pump power requirements are dependant upon the developed head across and the flow through the pumps. Due to similar primary hydraulic resistance between the two SG designs, the impact of the ROTSGs upon the power requirements and current draw of the reactor coolant pumps is negligible [Ref. 5].

Emergency Core Cooling Systems
The various ANO-1 emergency core cooling systems (ECCS) were reviewed for impact due to the ROTSGs. These systems include the high pressure injection (HPI) system, the low pressure injection (LPI) system, core flood (CF) system, and the reactor building spray (BS) system. These mechanical systems are necessary to mitigate the consequences of postulated accidents.

There are no physical changes to any of the ECCS systems due to the ROTSG installation. The operating system pressures and temperatures remain within the piping design values for these systems after installation of the ROTSGs. Following a design basis accident (DBA), the containment sump temperatures and levels are unaffected due to the ROTSGs. The net positive suction head available to the ECCS systems for post-accident recirculation is not adversely impacted due to the ROTSGs [Refs. 10, 31]. There are no required changes to the CF tank inventory requirements or injection setpoints [Ref. 29].

As discussed in Reference 36, the ROTSG peak containment pressure after a DBA is bound by the current OTSG analysis. Therefore, the BS system will continue to meet design flow requirements [Ref. 10].

Emergency Feedwater System
There are no physical modifications to the EFW system due to the ROTSG installation (other than the header and riser piping into the nozzles on the ROTSGs). The design, qualification, and operating characteristics of the EFW piping are acceptable with the ROTSGs.

The ROTSGs do not adversely affect the loss of feedwater accident analysis or the EFW sizing requirements. The OTSGs originally utilized seven EFW nozzles, however, one of the nozzles has been blanked off on both generators, yielding six viable EFW flowpaths into the tube bundle of each generator. The ROTSGs utilize all seven EFW nozzles. The effect of the additional flowpath of EFW into the replacement SGs is minor [Ref. 17]. The additional riser and nozzle has no significant impact upon the EFW system resistance. The required power to drive the equivalent EFW fluid is the same for the ROTSGs as for the OTSGs. The EFW minimum flow requirements are unchanged with the ROTSGs. The maximum EFW flow to a given SG is currently limited at 2000 gpm to limit the fatigue usage factors of the tubes in the vicinity of the EFW nozzles. This limit is bounding and acceptable for use for the ROTSGs.

Main Steam System
With one exception, there are no physical modifications to any of the components of the main steam system within the scope of ER-ANO-2002-1381-000. The one exception is that the ROTSG steam outlet nozzle elbows have a bend radius of 27" whereas the OTSG steam outlet nozzle elbows have a bend radius of 36". This difference, as discussed above, is due to the addition of the flow restricting venturis. The impact of these elbows on the existing piping and support has been evaluated and determined to be acceptable [Refs. 60, 61].

The higher secondary design pressure and temperature of the ROTSGs do not invalidate the design pressures and temperatures of the main steam system [Ref. 54]. All components within the main steam system will continue to operate within their design bases with the ROTSGs [Ref. 26].

For the limiting main steam line break in Room 170, installation of the ROTSGs will result in slightly worse steam conditions and resultant accident environments (temperatures, pressures and flooding potential). Refer to the Engineering Programs discussion for a description of the environmental qualification impact due to this line break.

Turbine Cycle Systems
The turbine cycle systems were evaluated for impact due to the installation and operation of the ROTSGs. The evaluations considered the following ANO-1 mechanical systems

- Turbine Generator System (Mechanical)
- Miscellaneous Turbine Drains
- Reheat Steam System
- Extraction Steam System
- Heater Drains
- Heater Vents
- Condenser
- Condenser Vacuum System
- Condensate System
- Main Feedwater System
- Circulating Water System

There are no physical modifications to any of these systems within the scope of ER-ANO-2002-1381-000. The higher secondary design pressure and temperature of the ROTSGs do not invalidate the design pressures and temperatures of the turbine cycle systems. The impact of the increase in main steam temperature on expected turbine cycle parameters was characterized in a series of PEPSE turbine cycle heat balances [Ref. 51]. Certain turbine cycle lines will experience slight changes to their current operating pressures and temperatures. However, these changes remain within the original design criteria of the respective lines. Therefore, the increase in main steam temperature has an insignificant effect upon the turbine cycle systems and all systems remain within their design bases and will continue to fulfill their design function. [Ref. 8]

The condenser heat load at 100% power is not significantly affected by the installation of the ROTSGs. Therefore, the Environmental Screening question regarding a possible increase in the discharge temperature to the lake is checked "no". [Ref. 8]

Decay Heat System
The ROTSG has a dry weight reduction of approximately 67 tons of metal from the OTSG but a full load secondary inventory increase of approximately 2500 lbm per generator. The resulting decrease in metal mass and primary inventory and increase in secondary inventory produce a change in the full load sensible heat of the generators. However, the decrease in sensible heat due to the smaller primary inventory and ROTSG metal mass offset any increase in sensible heat due to the slight increase in secondary inventory. Therefore, the sensible heating value of the RCS (metal and inventory) upon installation of the ROTSG remains bounded by the existing values. Therefore, the load to the service water system from the decay heat coolers and eventually, the ultimate heat sink (Lake Dardanelle or the emergency cooling pond) during cooldown is bounded by the original analyses.

The time to boil, time to core uncovery, and time to RCS repressurization upon loss of decay heat removal is dependant upon the reactor power history, time since shutdown, and RCS configuration. As described above, the installation of the ROTSGs will result in a slight decrease in primary volume. However, this slight decrease has a negligible effect on time to boil, time to core uncovery, and time to RCS repressurization; thus operation with the ROTSGs, relative to these parameters, remains bounded by current analyses [Ref. 31].

Makeup and Chemical Addition Systems
There are no physical changes made to the makeup or the chemical addition systems within the scope of ER-ANO-2002-1381-000.

The normal letdown flow permits recirculation of one RCS volume per day through the purification demineralizers and makeup filters. The total RCS volume is slightly decreasing due to the ROTSGs; therefore, this design goal for the purification system remains bounding. Similarly, the makeup tank remains appropriately sized to accommodate letdown, makeup, and volume changes, i.e., there is no impact to its functional capabilities.

The small changes in total primary and secondary volume will not prevent the chemical addition system from being able to control the pH and oxygen content of the primary and secondary systems.

NaOH addition raises the pH of the borated water from the borated water storage tank into a range that better facilitates the removal of iodine. The contents of the NaOH tank are proportioned so the proper amount of NaOH is drawn into the Reactor Building spray fluid for pH control. With the installation of the ROTSGs, the reactor coolant volume will slightly decrease, however, the NaOH concentration limits analysis remains bounding. Therefore, the NaOH tank concentration requirements remain unchanged with the addition of the ROTSGs.

Nuclear Safety Analysis Discipline

The nuclear safety analysis discipline review includes an extensive review of the FSAR Chapter 14 safety analyses and ECCS performance requirements as well as other related analyses important to design and licensing requirements. The effects of the ROTSG design differences upon the safety and ECCS performance analyses were evaluated both qualitatively and through reanalysis. These evaluations are detailed in ER-ANO-2002-1381-005. The objective of each evaluation is to demonstrate that the analysis presented in the ANO-1 FSAR with the OTSGs was bounding (quantitatively), or that the plant response, evaluated using ROTSG parameters, continues to meet all acceptance criteria following SG replacement (reanalysis). All events were determined to be bounding for operation with the ROTSGs. Two events required reanalysis: main steam line break and turbine trip. A brief summary of these events is discussed below.

Main Steam Line Break
The main steam line break (MSLB) analyses are discussed in Chapter 14 of the ANO-1 SAR. The MSLB event has the potential for impact due to the higher main steam temperature and larger secondary inventory of the ROTSGs. For core response, the steam line break area and the SG initial inventory are key parameters in the steam line break analysis. The steam outlet nozzles in the ROTSGs each contain flow-limiting venturis. The MSLB reanalysis showed that the additional ROTSG inventory and the flow-limiting venturis served to slow down the SG depressurization and extend the transient. Another effect of the venturis is a rapid depressurization of the steam lines once the break opens. The rapid depressurization of the intact-side steam line results in a much earlier lowpressure EFIC signal. This produces an earlier closure of the main feedwater isolation valve and the main steam isolation valve which lessens the severity of the overcooling accident. The revised core response analysis of the steam line break event showed that the analysis results are not adversely affected by the ROTSGs [Ref. 32].

The SG tubes must not fail as a result of the loss of secondary side pressure or the resultant temperature gradients. Structural analyses that consider various transient loadings have been performed to determine allowable flaw configurations based on the structural integrity requirements outlined in the current version of the Nuclear Energy Institute (NEI) initiative NEI 97-06, Steam Generator Program Guidelines. The pressurizer surge line break (PSLB) is the limiting design break for tube loads and bounds the conditions resulting from a main steam line break. Reanalysis has demonstrated that the PSLB tube loads do not result in tube failure; thus, the SG tubes will not fail as a result of a steam line break [Ref. 33]. Further evaluations conclude that SG tube failures will not occur as a result of a the high cross-flow steam velocities during a steam line break accident with the ROTSGs [Ref. 15].

Main Steam Line Break Containment Response
The containment response was not recalculated, but was evaluated using the results for the reanalysis of the main steam line break (MSLB) accident which was performed with the inclusion of the ROTSG in the RELAP5/MOD2-B&W input deck. The RELAP5/MOD2-B&W code was utilized to predict the plant system

response and mass and energy release rates to the reactor building from the consequent blowdown of the ruptured steam line. The RELAP5/MOD2-B&W code has been approved by the NRC for predicting the plant system response to non-LOCA accidents for B&W-designed plants with once-through SG designs.

Consistent with the MSLB Upper Level Document, the mass and energy release rates were obtained from the core response analysis [Ref. 55]. It was demonstrated that the ROTSG specific blowdown integrated mass and energy data is bounded by the blowdown integrated mass and energy data utilized in the reactor building pressure analysis discussed in Section 14.2.2.1.5 of the FSAR [Ref. 36]. Therefore, the long term containment response from a MSLB with the ROTSG is bounded by the OTSG MSLB analysis of record documented in Section 14.2.2.1.5 of the FSAR.

The ROTSG MSLB enthalpy values imply that there will be higher superheated conditions in the reactor building during a MSLB with the ROTSG design. The more superheated steam exiting the ROTSG will drive the reactor building to a higher containment atmosphere temperature in comparison to the OTSG analysis of record. Since there is always a superheated temperature spike following the initiation of an OTSG MSLB, EOI has incorporated the NRC position from NUREG-0458 into the ANO-1 design basis [Ref. 53]. The NRC position is that because initial superheating of the containment atmosphere causes a temperature spike of a short duration, thermal lag of the equipment inside containment ensures that the equipment temperature will not increase significantly during the superheated period. Consequently, the initial temperature peak does not define operating limits on any SSC and the long-term temperature response of the atmosphere, which is essentially the saturation temperature at the containment pressure, dominates the temperature response of equipment inside containment. Because the peak MSLB pressure is below the peak LOCA pressure, the equipment qualification envelope is defined by LOCA.

Turbine Trip
A turbine trip is limiting in terms of the maximum secondary pressure and is summarized in the ANO-1 Overpressure Protection Report [Ref. 22]. This analysis supports Tech Spec limits for the allowable number of MSSVs to be out of service based on operating power level as defined in the bases of Tech Spec 3.7.1. The reactor protection criterion as defined by the ASME code states that the rated capacity of the pressure-relieving devices shall be sufficient to prevent a rise in pressure within the vessels which they protect of more than 10% above the design pressure. A turbine trip event produces a rapid increase in secondary pressure. The reanalysis of the turbine trip with the ROTSGs at 114% of 2568 MWt showed a lower peak secondary pressure than the analysis of record for the OTSG at 102% of 2568 MWt. The reduction in peak secondary pressure was a result of an improved MSSV analytical model and is not attributed to the ROTSG. The peak primary pressure predicted for the turbine trip with the ROTSGs increased, but this is attributed to the higher initial core power modeled in the analysis. Both primary and secondary pressures remained below the ASME code criterion of 10% above the vessel design pressures. The revised system analysis of the turbine trip event demonstrated that the analysis results are not adversely affected by the ROTSGs [Ref. 35].

Other Safety Analyses Evaluated to Support OTSG Replacement

In addition to the safety analyses which were analyzed to support OTSG replacement, all of the Chapter 14 events and several other analyses important to nuclear safety and design requirements were evaluated. The analyses evaluated are listed below:

ANO-1 SAR Chapter 14 Analyses

- Uncompensated Operating Reactivity Changes
- Startup Accident
- Rod Withdrawal Accident at Rated Power Operation
- Moderator Dilution Accident
- Cold Water Accident
- Loss of Coolant Flow
- Stuck-in, Stuck-out, Dropped Control Rod
- Loss of Electric Power
- Turbine Overspeed
- Fuel Loading Errors
- Steam Line Failure
- Steam Generator Tube Failure
- Fuel Handling Accident

Control Rod Ejection Accident
Loss of Coolant Accident
Maximum Hypothetical Accident
Waste Gas Tank Rupture
Emergency Feedwater System Sizing

Other Design and Licensing Analysis/ Issues

Anticipated Transient Without Scram
Core Lift
End of Cycle T_{ave} Maneuver
Decay Heat Removal Capability in Natural Circulation
Overpressure Protection
Low Temperature Overpressure Protection (LTOP)
Station Blackout
Small Steam Line Break (PSC 7-78)
Steam Generator Tube Loads
Reactor Coolant Pump Two Minute Trip Criteria (PSC 2-00)
Boron Precipitation (PSC 2-91)
Boron Dilution (PSC 1-95)
Post-LOCA Hydrogen Generation
Venturi Critical Flow
Containment Evaluations
SG Levels to Protect Steam Line Break During Heatup/Cooldown

SAR Chapter 14 Dose Consequences
The installation and operation of the ROTSGs required the re-evaluation of the design basis accidents and transients to determine if the new design features associated with the ROTSGs (such as flow restricting venturis, decrease in primary inventory, increase in secondary inventory, decrease in overall SG mass, etc.) would have an adverse effect upon the dose consequences of these accidents and transients. The evaluations determined that there is no increase in the dose consequences of any accident previously evaluated in the FSAR [Ref. 37].

Piping Discipline

The piping discipline review includes an extensive evaluation of the impact of the ROTSGs upon the stress analyses for the main coolant loop piping and supports, RCS tributary piping and supports, and relevant secondary systems piping and supports. This review demonstrates that the design and licensing requirements for the piping, components, and supports remain acceptable for use with the ROTSGs. The evaluations concluded that the loads acting upon the piping, components, and supports either do not increase above the design basis loads when the ROTSGs are introduced into the RCS or the stresses present after the ROTSG introduction continue to meet the allowable stresses dictated by the applicable design codes.

Main Coolant Loop Piping
The stresses in the RCS primary piping and the surge line were evaluated for the ROTSG loop analysis utilizing LBB criteria. Additionally, the primary nozzles for the reactor vessel and reactor coolant pumps were evaluated for acceptability with the ROTSGs. The analysis demonstrated that the primary RCS piping is qualified for use with the ROTSGs in place. The stress analysis was completed using B31.7, 1968 code year as this is the code of record for ANO-1 [Refs. 38, 41]. The primary loop component nozzles were qualified using the load comparison method [Ref. 38].

The surge line was qualified separately from the main coolant loop piping. The surge line undergoes many severe thermal transients that can not be considered in a simplified analysis. Therefore the surge line was given a detailed analysis considering all thermal stratification transients previously considered for the surge line [Ref. 18]. All primary stresses on the surge line piping and elbows were determined to remain bounded by the current analyses. As the primary operating parameters of the ROTSGs are essentially equivalent to the OTSGs, the secondary (thermal expansion) stresses were determined to be unchanged with ROTSG operation. In addition, no changes in fatigue of the surge line components due to the ROTSGs was identified [Ref. 38]. However, the review determined that two surge line support snubbers were undersized. These will be replaced with higher capacity snubbers in 1R19 (not a part of this ER).

RCS Main Coolant Loop and Component Supports
The SG, reactor coolant pump, pressurizer and reactor vessel vertical and lateral supports and embeds were demonstrated acceptable for the ROTSG loop analysis. The RCS loop analysis was revised taking credit for LBB methodology. When possible the primary loop component supports are qualified using the load comparison method. he maximum possible load for a type of support was found from the ROTSG analysis and was then compared to the appropriate allowable load from the applicable stress report or manufacturer's catalog. If the support component could not be shown acceptable by load comparison, then the affected support required further evaluation to determine acceptability and stresses were conservatively recalculated to account for the change in load. All supports were demonstrated to be acceptable and remain qualified using this method [Ref. 38].

RCS Tributary Piping
Piping attached to the main coolant loop piping (namely, surge line, decay heat line, spray line, HPI and HPI/makeup line, pressure taps and flow meter lines, vent and drain lines, and the letdown/drain nozzle) was evaluated for structural impact due to operation with the ROTSGs. There are no physical modifications to any of the RCS tributary piping as a result of ER-ANO-2002-1381-000 (the removal of the primary drain piping is included in ER-ANO-2002-1078-014, "2" and Under Piping and Supports"). The displacements from the ROTSG analysis are acceptable if the total ROTSG displacement value, either thermal or seismic is less than 1/16 inch, if the ROTSG displacement value is less than the previous analysis, or if the ROTSG displacement value is not more than 5% greater than that of the previous analysis. The piping displacement analysis demonstrated that the majority of the attachment locations required no further evaluation and remain qualified for the ROTSGs [Ref. 38]. Further evaluation was required for the following attached piping systems:

Loop A2 and B1 2½" HPI lines
Loop A2 1" level indication line
Loop A2 2½" spray line
Loop A and B ROTSG 1½" temperature sensing line
Loop A and B ROTSG 2" tubesheet drain line
Loop A2 and B1 RCP center of gravity of motor location
Loop A2 and B1 RCP hanger attachment location

Secondary Systems Piping
Secondary systems piping (main steam, main feedwater, and emergency feedwater) were also evaluated to determine if the ROTSGs adversely affect these piping systems from a structural perspective. All piping systems were evaluated between the reactor building (RB) penetration and the respective nozzles and included the MFW and EFW support changes that are detailed in ER-ANO-2002-1078-013; these evaluations demonstrated continued code compliance after installation of the ROTSGs (Refs. 56, 57, 58, 59, 60, 61). The analyses were performed to the current code of record with the exception to the replacement main and emergency feedwater header ring header and riser piping. The MFW and EFW ring headers and riser piping were qualified to ASME Section III, Subsection NC, 1989 Edition as required per Reference 47. A reconciliation for the different codes has been performed [Ref. 7].

Major RCS Components
There are no physical modifications to any of the major RCS components (with the exception of the SGs and their associated piping connections) within the scope of ER-ANO-2002-1381-000. The pressurizer, reactor vessel, and reactor coolant pumps are physically unchanged. Evaluations compared the resulting loads on the components (ROTSGs installed) against the existing design loads taken from the component stress reports. All ROTSG-driven loads were demonstrated to be bounded by existing loads or the resulting stresses were successfully recalculated and shown less than code allowables [Refs. 39, 40, 42, 43]. The reactor vessel internals were evaluated for seismic loads due to the ROTSGs and shown acceptable. Leak-before-break (LBB) has been utilized with ROTSG reanalysis. As there are no significant differences in the primary side characteristics, high energy line break loads are not affected by the SG replacement. Therefore, evaluations demonstrate that the pressurizer, reactor vessel, reactor vessel internals, and reactor coolant pumps will maintain their structural integrity for all loading conditions, including normal operating and upset, emergency, and faulted conditions. These evaluations concluded that the design, qualification, and expected operating characteristics of the major RCS components are acceptable for the ROTSGs at 100% power (i.e., 2568 MWt).

In addition, an evaluation of the design of the ROTSGs, their supports, and their internals was performed. The evaluations demonstrated that the ROTSGs meet all ASME code requirements. Existing support (upper lateral support) evaluations were re-visited to demonstrate acceptability for use with the ROTSGs [Ref. 15].

Leak-Before-Break
The successful application of leak-before-break (LBB) to the main RCS piping is described in report BAW-1847, "The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS", Revision 1. This report provides the technical basis for evaluating postulated flaw growth in the main RCS piping under normal plus faulted loading conditions. LBB LOCA loadings are considered for the faulted analyses of fuel assemblies to justify pipe whip restraints being no longer required on the main RCS piping and removed as needed to facilitate maintenance or other activities. In addition, the analysis of reactor building internal pressure differentials following a LOCA applies LBB in the selection of breaks to be analyzed. LBB criteria is related to crack growth in the primary coolant piping and the containment building leak detection systems. The ROTSGs have no impact upon the LBB methodology or the leak detection systems at ANO-1. However, the RCS Loop analysis was revised to take credit for LBB methodology as part of the ROTSG design. Both normal operating and faulted loads were determined to be enveloped by the existing loads [Ref. 38]. Therefore, the ANO-1 main coolant loop piping is acceptable for LBB with the ROTSGs installed.

To facilitate replacement of the OTSGs, an approximate 22.5° section of each cold leg elbow will be removed with the OTSG. This section of cold leg piping will be replaced with a forged elbow section that is integral to the ROTSG. These new cold leg elbows were evaluated and determined to be justified for LBB application. Likewise, a 135° section of the hot leg elbow will be replaced. The LBB qualification of the new hot leg elbows and the new welds that will connect ROTSGs to the RCS piping are addressed in ER-ANO-2002-1078-015. FSAR updates were included in the LBD Change for ER-ANO-2002-1078-015 and in ER-ANO-2002-1381-000 to document that LBB was used in the pipe rupture analysis (Ref. 38) to determine the appropriate pipe rupture locations, and to eliminate the dynamic loads from a hot leg or cold leg break on the steam generator upper lateral restraints and lower base support.

Civil Discipline

The civil discipline review (Ref.13) addressed the impacts due to the ROTSGs upon the civil and structural items associated with ANO-1. These items include concerns such as component supports, high energy line breaks, and a Reactor Building evaluation. The evaluations determined that there are no adverse impacts due to the ROTSGs.

Electrical / Instrumentation and Control Discipline

The electrical / instrumentation and control (E/I&C) discipline review (Ref. 16) includes an extensive review of the ROTSGs impact upon E/I&C plant SSCs. As described above, an expert panel of experienced engineers knowledgeable in ANO-1 SSCs and plant operations selected appropriate systems as those which could potentially be adversely affected by changes to system operating parameters due to operation with the ROTSGs. These systems were reviewed through the various daughter ER responses. A brief summary of the review results is discussed below. These ER responses determined that there are no physical plant changes associated with these systems that are required to support the ROTSGs and plant operation. All tap elevations, instrument hardware, and control room indications with the exception of LT-2621 as discussed below remain unchanged within the scope of ER-ANO-2002-1381-000.

Integrated Control System
As part of the E/I&C discipline review, the Integrated Control System (ICS) was evaluated for potential impact due to the operation of the ROTSGs. There are no physical modifications to any of the ICS components as a result of the ROTSG installation. All components, power supplies, and instrumentation will remain in their original configuration with the original equipment. The ICS performs three functions. First, for reactor power levels of 15% to 100%, plant equipment is automatically controlled to match reactor heat input with SG heat transfer and turbine heat removal. Second, below 15% reactor power, plant equipment is automatically controlled to match SG heat transfer and heat removal with the manually controlled reactor power. Lastly, the ICS gives the operator the capability to manually control the equipment normally controlled by the ICS [Ref. 30].

The evaluation determined that an extremely close match between the OTSG and the ROTSG operating parameters exists during normal full-power steady-state conditions. Because the form, fit and function of the ROTSGs are nearly identical to the OTSGs, the function of the ICS System (and its design bases), the interfaces to other systems, and applicable calculations are not adversely impacted by the installation of the ROTSGs. All setpoints and alarms remain unchanged with the ROTSGs. The evaluation demonstrated that the installation of the ROTSGs will have only minor impacts upon the configuration, operation, performance, maintenance, inspection, or testing of the ICS. [Ref. 30].

Emergency Feedwater Initiation and Control
An evaluation of the impact of the ROTSGs upon the Emergency Feedwater Initiation and Control (EFIC) system was conducted [Ref. 50]. All EFIC power supplies, cabling, components and instrumentation with the exception of the EFIC level taps will remain in their original configuration with the original equipment. However, some EFIC level taps will be moved azimuthally. Since the elevations of all taps for both the Low Level Range and the High Level Range measurements are unchanged, there will be no impact to the operation or readings of the EFIC level transmitters. In addition, replacing the SGs was determined to have no impact on the instrument ranges of these level transmitters for the EFIC system or the Post Accident Monitoring (PAM) indicators or recorder displays. Additionally, replacing the SGs was determined to have no impact on the instrument ranges of the pressure transmitters for the EFIC system or the PAM indicators or recorder displays. Since the EFIC circuits, logic, algorithms, and setpoints remain unchanged, the time response of the EFIC channels also remain the same so as to not impact any equipment actuation time requirements. Comparison of the control setpoints for the OTSG and the ROTSG show that there aren't any requirements to change the level, pressure, and fill rate setpoints for the SG upgrade. The ROTSG design will not impact the natural circulation limit of 312 inches, the reflux boil limit of 378 inches, nor require any change to the EFW valves or pumps. Also, the EFIC low level control setpoint (31") will not be required to be changed.

The EFIC operator interfaces will remain unchanged with the implementation of the ROTSG. Operation of switches and observation of annunciators will remain the same and don't require any hardware change outs [Ref. 50].

[Note: One EFIC level transmitter (LT-2621) currently in SG24B has a tap that extends through the SG shroud into the tubesheet region. This tap will go only into the downcomer region in the SG24B, which will match the existing configuration of the other EFIC low level transmitters. See plant computer system discussion below for impacts.]

Reactor Protection System
The effects of the installation of the ROTSGs at ANO-1 upon the reactor protection system (RPS) were evaluated based on 2568 MWt. All components, power supplies, and instrumentation will remain in their original configuration with the original equipment. The individual automatic trip functions of the RPS are overpower trip, power trip based on imbalance and flow functions, reactor outlet temperature trip, pressure-temperature trip, reactor pressure trips, power/reactor coolant pumps trip, reactor building pressure trip, and anticipatory reactor trip system (ARTS). Overpower is detected using the linear neutron flux signal. The installation of the ROTSGs will have no impact on the neutron flux levels. The inputs used for the power trip based on imbalance and flow are the signals for neutron flux and reactor coolant flow. As stated previously, neutron flux is unchanged. Reactor coolant flow will increase slightly from current values, however, the small increase in coolant flow is insignificant and will not impact this RPS trip function. Additionally, the power/imbalance/flow trip setpoints are verified or changed at the beginning of each fuel cycle, and are based upon the core operating limits report (COLR) for the fuel load, so any change in coolant flow will be accommodated. Reactor outlet temperature trip uses the signal for reactor outlet temperature. Because DNBR increases due to increased RCS flow, the high temperature trip setpoint remains bounded for ROTSG operations. Pressure-temperature trip uses the signals for RCS pressure and reactor temperature. The pressure-temperature trip setpoint is also verified at the beginning of each fuel cycle. Reactor pressure trip uses the signal for RCS pressure. As stated previously, the ROTSG primary side design pressure and temperature are the same as the OSTG. Because of the close similarity of the ROTSGs to the OTSGs, the relationship between allowable power and the number of operating pumps is unchanged. The reactor building pressure trip uses the signals for reactor building pressure. As the OTSG containment pressure and temperature response to the limiting accident scenarios has been shown to bound the ROTSG response, the containment design pressure remains unchanged due to the SG replacement. The ARTS uses detection of turbine trip or a loss of both main feedwater pumps to initiate a reactor trip. Neither condition is impacted by the ROTSGs.

The evaluation demonstrated that the installation of the ROTSGs will have no impact upon the configuration, operation, performance, maintenance, inspection, or testing of the RPS SSCs [Ref. 64].

Engineered Safeguards Actuation System
There are no physical modifications to any of the ESAS components as a result of the ROTSG installation. All components, power supplies, and instrumentation will remain in their original configuration with the original equipment. The function of the system will remain the same; upon initiation of ESAS, reliable actuation of engineered safeguards will occur as before. The setpoints and bases related to the ESAS System were reviewed for impact. The evaluation demonstrated that the installation of the ROTSGs will have no impact upon the configuration, operation, performance, maintenance, inspection, or testing of the ESAS [Ref. 52].

Plant Computer System
A review of the Plant Computer System for ROTSG impact identified no changes to the Plant Computer System; operator interfaces will remain unchanged with the implementation of the ROTSG. No computer point's range associated with ROTSG affected process parameters will have to be re-spanned. The OTSG level tap associated with SG 24B, Channel D, Low Level Transmitter, LT-2621 is different from the other OTSG level taps in that it extends through the shroud into the tubesheet, while the other taps stop in the downcomer region. With the ROTSG installed, the Plant Computer will no longer be required to apply a correction factor to EFIC channel D low level initiate (L2621-C) in support of the EFIC cross channel-check of OPS-A22, since its configuration will not be different from the other EFIC Low Level Transmitters. A software change was initiated to address this. The Plant Computer System was determined to remain acceptable for use with the ROTSGs [Ref. 62].

Others
In addition to the above systems, other E/I&C systems were evaluated to ensure acceptable operation with the replacement SGs. These systems include Diverse Reactor Overpressure, Emergency Diesel Generator, 6.9 kV Switchgear, Annunciator, Nuclear Instrumentation, Non-Nuclear Instrumentation, Safety Parameter Display System, Main, Unit, Auxiliary, and Startup Transformers, Isophase Bus, Isophase Bus Cooling, Smart Automatic Signal Selector, and Vibration Loose Parts Monitoring systems. Each of these systems was demonstrated to remain within their design bases and to function satisfactorily with ROTSG operation. Refer to Reference 16 for more information regarding these system evaluations.

<u>Engineering Programs</u>

An evaluation was completed to assess the impact of the ROTSG installation and operation on the various ANO-1 plant programs (Ref. 6). The impact, if any, on the ANO-1 plant programs was assessed by evaluating the effects of changing SG parameters on the relevant ANO-1 plant program. As described previously, major changes in SG parameters with installation of the ROTSGs used for this evaluation were documented in the Parameter Impact List (PIL) (Ref. 20). Programs identified as having a potential for impact due to the ROTSGs are further evaluated below.

Seismic Qualification
As the ROTSGs have different weight and center of gravity characteristics than the OTSGs, ROTSG installation will impact the seismic qualification of plant SSCs at ANO-1. Reference 46 evaluates the seismic qualification of the replacement steam generators. The piping discipline review (Ref.14) evaluates the seismic qualification of all attached piping, mechanical equipment (e.g., valves), and electrical equipment (e.g., transmitters).

Flow Accelerated Corrosion
In principle, installation of the ROTSGs could negatively affect the Flow Accelerated Corrosion (FAC) program in two ways: it could change the classification of a piping segment from non-susceptible to susceptible, or it could increase the susceptibility of a piping segment that is currently classified as susceptible. The plant systems that could be affected by FAC are limited to the reactor coolant system, SGs, main steam system, and turbine cycle mechanical systems. The reactor coolant system is fabricated from materials that are not susceptible to FAC (e.g., stainless steel). The ROTSGs are evaluated for FAC in (Ref. 49). Steam temperatures in the main steam system are higher with the ROTSGs and are bounded by existing FAC evaluations. Similarly, the steam qualities in turbine cycle mechanical systems are higher with the ROTSGs and are bounded by existing evaluations. As such, line segments currently classified as non-susceptible to FAC will be unaffected by installation of the ROTSGs.

Piping segments that are presently identified as susceptible to FAC could be affected by installation of the ROTSGs. With no changes in materials or geometry in the main steam or the turbine cycle mechanical systems, FAC can only be negatively affected by adverse changes in fluid process conditions (i.e., temperature, chemistry, velocity, and moisture content). However, the ROTSGs will supply equivalent superheat at loads less than about 70% reactor thermal power, and higher superheat at loads greater than about 70% reactor thermal power. Additionally, the main steam flow rates at full load will be reduced about 0.4% with installation of the ROTSGs. Therefore, the susceptibility to FAC of the main steam system with installation of the ROTSGs will be bounded by operation with the OTSGs. For the turbine cycle mechanical systems, the effects of any temperature, velocity, or moisture content changes on FAC susceptibility with installation of the ROTSGs is negligible or bounded by the current conditions. Therefore, the FAC program was determined to be unchanged due to the installation of the ROTSGs [Ref. 6].

Motor Operated Valve Program
The effects of the ROTSG upon the MOV program was evaluated. Installation of the ROTSGs will not add, delete, replace or affect any MOV, the electrical system or hardware, or the control circuitry of any MOV in the Generic Letter 89-10 program. Other than the limiting Main Steam Line Break, the accident environments (temperatures and pressures) outside containment are unaffected by installation of the ROTSGs. For the limiting Main Steam Line Break in Room 170, CR-ANO-1-2004-00104 addresses the potential non-conservative steam conditions assumed in CALC-83-D-1034-03. The Operability Evaluation for this CR concluded that the operability of components in Room 170 would not be impacted. CALC-83-D-1034-03 will be updated to bound steam conditions for the OTSG and ROTSG. Installation of the ROTSGs will not change the operational state or stroke time of any MOV in the Generic Letter 89-10 program. The review concluded that no changes are required for the installation of the ROTSGs that would affect the program content of the ANO-1 MOV program [Ref. 6].

Containment Leak Rate Testing
In accordance with 10 CFR 50, Appendix J, containment leak rate testing should be based on the calculated peak containment internal pressure related to the design basis accident and specified either in the Technical Specifications or associated bases. As described in Reference 36, the peak containment internal pressure for the limiting design basis accident will be unaffected by installation of the ROTSGs. Since the calculated peak containment internal pressure is unaffected by installation of the ROTSGs, the ANO-1 Appendix J testing program is also unaffected by installation of the ROTSGs.

Appendix R: Fire Protection
An evaluation was completed to assess the impact of the ROTSG installation on the ANO-1 fire protection program. Impacts were assessed based on a review of ANO-1 design basis documentation, licensing basis documentation, and ROTSG design documentation. This evaluation determined that the ROTSGs will have a negligible impact on the ANO-1 Fire Protection Program.

Environmental Qualification
The environmental qualification program was assessed to determine the impact, if any, of the ROTSG installation. The normal operational environment (which includes temperature, pressure, humidity, radiation, submergence, and vibration) for inside and outside the containment building were determined to be not adversely affected due to operation with the ROTSGs.

For accident environments, containment pressure and temperature excursions are bound by LOCAs and MSLBs. However, the current OTSG licensing position states that the worst-case high energy line break (HELB) conditions inside containment need only be based on LOCA conditions. Therefore, the worst-case HELB inside containment (from an EQ perspective) is a design basis LOCA. As discussed in Reference 36, the LOCA accident environments inside containment with the ROTSGs will be no worse than the existing OTSG LOCA profiles. For the limiting Main Steam Line Break in Room 170, CR-ANO-1-2004-00104 addresses the potential non-conservative steam conditions assumed in CALC-83-D-1034-03. The Operability Evaluation for this CR concluded that the operability of components in Room 170 would not be impacted. CALC-83-D-1034-03 will be updated to bound steam conditions for the OTSG and ROTSG. Other than the limiting main steam line break, the accident environments (temperatures and pressures) outside containment are unaffected by installation of the ROTSGs. The impact of the ROTSG installation on the post-LOCA sump pH and humidity was determined to be negligible. The accident radiation doses both inside and outside containment were determined to be unaffected by installation of the ROTSGs. The ROTSG worst-case submergence inside containment was determined to be bounded by the current predicted maximum.

Chemistry
The water chemistry requirements of the ROTSGs are essentially identical to the OTSGs. Electric Power Research Institute (EPRI) water chemistry guidelines for both primary and secondary water requirements are acceptable for the ROTSGs [Ref. 44]. During the initial plant startup following replacement, metal release rates, especially for nickel, may be higher than equilibrium release rates. Industry experience with SG replacements has noted, in several cases, increased nickel releases during shutdowns and/or flux depressions in high duty cores following replacement. These observations have been difficult to assess due to other simultaneous changes in plant operations (e.g. core uprate, chemistry control). In other cases, post-replacement operation has been relatively problem-free. Nonetheless, the potential for greater crud deposition on the fuel raises concerns regarding elevated dose rates and impacts of nickel-based corrosion products on fuel performance. As previously discussed, a zinc injection system has been installed (Ref. 45) and had been injecting low levels of zinc into the RCS. According to AREVA evaluations, zinc injection will mitigate the susceptibility to nickel

releases and fuel AOA (Axial Offset Anomaly). Zinc injection is expected to be effective even if it has not been injected for the last few weeks of Cycle 19 and the first few weeks of Cycle 20 due to the slow buildup of nickel from the ROTSGs' tubes and the residual zinc remaining in the RCS (Ref. 80, 81).

References

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27. CALC-021381E101-09, ANO-1 EOTSG Design Transients, Revision 0.
28. CALC-021381E101-16, ANO-1 EOTSG Primary Hydraulics, Revision 0.
29. CALC-021381E101-134, ANO-1 EOTSG Impact Screening on Core Flood System, Revision 0.
30. CALC-021381E101-184, ANO-1 EOTSG Impact Screening on Integrated Control System, Revision 0.
31. CALC-021381E101-177, ANO-1 EOTSG Impact Screening on Low Pressure Injection System, Revision 0.
32. CALC-021381E101-306, ANO-1 Main Steam Line Break for Core Response, Revision 0.
33. CALC-021381E101-308, ANO EOTSG PSLB Tube Load Evaluation, Revision 0.
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54. CALC-021381E101-132, ANO-1 EOTSG T/H Calculation on Impact of Tsteam, Revision 0.
55. ULD-1-TOP-02, Upper Level Document for the Main Steam Line Break Accident, Revision 1.
56. CALC-91-E-0016-107, Qualification of MFW Line Inside Containment from Penetration P3 to SG EA24A Train A, Revision 1, DRN 04-2302.
57. CALC-91-E-0016-108, Qualification of MFW Line Inside Containment from Penetration P4 to SG EA24B Train B, Revision 1, DRN 04-2303.
58. CALC-92-D-5021-01, Piping Analysis for EFW Piping Inside Containment for E24A, Revision 2, DRN 04-2304.
59. CALC-92-D-5021-02, Piping Analysis for EFW Piping Inside Containment for E24B, Revision 1, DRN 04-2305.
60. CALC-93-E-0050-01, Evaluation of Main Steam Piping Inside Containment from E24B to Penetration P2, Revision 1, DRN 04-2306.
61. CALC-93-E-0050-02, Evaluation of Main Steam Piping Inside Containment from E24A to Penetration P1, Revision 1, DRN 04-2307.
62. CALC-021381E101-158, ANO-1 EOTSG Impact Screening on Plant Computer System, Revision 0.
63. ER-ANO-2002-1381-013, Umbrella ER - Systems Review Process.
64. CALC-021381E101-185, ANO-1 EOTSG Impact Screening on Reactor Protection System, Revision 0.
65. CALC-021381E101-31, Tube-to-Tubesheet Weld Stress Analysis, Revision 0.
66. CALC-021381E101-62, ANO-1 EOTSG RCS Structural Model, Revision 0
67. CALC-021381E101-63, ANO-1 EOTSG RCS Structural Loading Analysis, Revision 0
68. CALC-021381E101-51, Design Parameters List
69. CALC-021381E101-05, Structural Analysis of Flawed Tubes
70. CALC-021381E101-49, ASME Code Summary Design Report, Revision 0
71. CALC-021381E101-98, ANO EOTSG Non-Linear Wear Analysis , Revision 1
72. CALC-021381E101-114, ANO EOTSG Tube FIV Summary Report, Revision 0
73. CALC-021381E101-102, ANO EOTSG Non-Linear FIV Analysis, Revision 0
74. CALC-021381E101-41, EOTSG Instrumentation Nozzle FIV Evaluation, Revision 0
75. CALC-021381E101-208, ANO-1 E-OTSG Orifice Plate Velocity Conditions, Revision 0
76. CALC-021381E101-227 ANO-1 EOTSG Startup Test Requirements and Acceptance Criteria
77. FANP Document 32-5058282-00, “ANO-1 Cycle 20 RCS Hydraulics,” March 2005.
78. FANP Document 38-5057088-01, "ANO-1 Cycle 20 Safety Analysis Groundrules, May 2005."
79. FANP Document 32-5063066-00, "ANO-1 Cycle 20 Lift, June 2005."
80. CALC-A1-NE-2005-001, "ANO-1 Cycle 20 Reload Report, August 2005."
81. **ANP letter, « LATER »(Open Item)**

Check the applicable review(s): **(Only the sections indicated must be included in the Review.)**

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-030)**	**Sections I, II, and IV required**

Preparer: Robert Sharp / **ORIGINAL SIGNED BY ROBERT SHARP** / ENS / SGT / 9-24-05
Name (print) / Signature / Company / Department / Date

Reviewer: Morris Byram / **ORIGINAL SIGNED BY MORRIS BYRAM** / EOI / SG-RVCH / 9-27-05
Name (print) / Signature / Company / Department / Date

Reviewer: Chris Davenport / **ORIGINAL SIGNED BY CHRIS DAVENPORT** / DP Eng / SG-RVCH / 9-28-05
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Reviewer: Saket Sharan / **ORIGINAL SIGNED BY SAKET SHARAN** / DP Eng / SG-RVCH / 9-28-05
Name (print) / Signature / Company / Department / Date

OSRC: J.R. Eichenberger / **ORIGINAL SIGNED BY J.R. EICHENBERGER** / 10-04-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	Unit 1 FSAR – Changes are identified in the LBD change attached to ERANO-2002-1381-000 and below.
TS Bases	☒	☐	TS Bases Section B3.7.1 is updated by the LBD change attached to ERANO-2002-1381-000.
Technical Requirements Manual	☒	☐	Sections 3.4.3 and B3.4.3
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☐ **Yes**
☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change <u>AND</u> initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. <u>Basis</u>

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

<u>Operating License/Technical Specifications</u>

The ANO-1 Operating License, Technical Specifications, Technical Specification Bases, and Technical Requirements Manual were reviewed to determine the impact from the proposed replacement of the OTSGs. The review of the Technical Specifications and Technical Specification Bases is documented in Reference 4. This review noted that Section 5.5.9, Steam Generator (SG) Tube Surveillance Program contained various items that were not applicable to the ROTSG design. The report further noted that EOI may desire to update the program to the latest standards outlined in NEI 97-06. The report also identified a reference change in the Technical Specification Bases Section 3.7.1. This change is included in the LBD change request.

By letter dated September 15, 2004, EOI submitted a License Amendment Request to the NRC to change the ANO-1 Steam Generator Tube Inservice Inspection Program. The proposed changes delete the current Technical Specification 5.5.9, Steam Generator (SG) Tube Surveillance Program and replace it with a new Section 5.5.9 titled "Steam Generator (SG) Program". Other changes and additions were also proposed consistent with the guidance in the industry initiative on NEI 97-06, Steam Generator Program Guidelines, and in the Technical Specification Task Force TSF 449, Draft Revision 3. The proposed changes to the ANO-1 Technical Specifications and Bases addressed all of the ROTSG-related changes identified in AREVA/FANP Document 51-5018858-00, "ANO Unit 1 Technical Specification Review". The requested changes were approved by the NRC on August 10, 2005 in Amendment 224 to the ANO-1 Facility Operating License.

<u>FSAR</u>

Each section of the ANO-1 FSAR has been reviewed to determine the effects of use of the ROTSGs. The steam generators are discussed in numerous other sections of the FSAR.

Two types of changes were identified for the ANO-1 FSAR text, tables and figures. One type is informational in nature: the affected sections describe the steam generator, the materials employed, and the thermal hydraulic characteristics. These areas required changes to Chapters 1, 4, and 10.

Sections 1.1, 1.8, and 10.4.9 identify B&W as the NSSS supplier. These sections will be revised to reflect that the replacement steam generators were supplied by AREVA/Framatome-ANP. References were added to sections 4.5 and 14.4.

The FSAR includes a number of references to the original OTSG design and qualification. These discussions will be revised or deleted in Sections 1.5.1, 1.10, 4.3.4.2, and 4.1.2.5.2, 4.1.2.6, 4.2.2.2, 4.3.12.2, 10.4.8 and A.7.1, and Figures 4.5 and 4.9 as they apply to replacement OTSGs. Figures 4-13 and 4-15 are deleted.

The FSAR includes a number of discussions of the ASME Code Section and Code Cases that were used for the original OTSGs. Sections 1.4.1, 4.1.2.3,4.1.3.1,4.2.1.2, 4.2.6, A.2, and Tables 1-1, 4-2, and 4-2A will be revised to reflect the Code Sections and Code Cases used in the ROTSGs and use of LBB criteria.

(continued)

The second type of ANO-1 FSAR modifications result from the safety evaluations performed to support use of the ROTSGs. Specifically, the safety evaluations in Sections 4.2 through 4.6 of the FSAR were reviewed and verified that the plant structural and thermal-hydraulic responses to upset and accident conditions with the ROTSGs are bounded by the original calculations or are within the applicable FSAR acceptance criteria. Sections 4.2 through 4.6 (4.2.6.3, 4.3.4.2, 4.3.6, 4.3.12.1,4.4.1, Tables 4-4, 4-8, 4-8A, 4-9, 4-12, 4-18, and 4-20) were revised as needed. Chapter 14 was revised to note that the current LOCA and non-LOCA analyses remain bounding for use of the ROTSGs. Wording was also added to sections 14.1 and 14.2 stating that the MSLB was reanalyzed and that the current licensing basis discussion is bounding.

Section 4.3.4.3 was revised to reference the use of NEI 97-06 consistent with technical specification 5.5.9.

Section 4.3.8 was updated to describe the new Turbine Trip analysis performed to support OTSG replacement.

In addition, clarification wording was added or changed in sections 1.10, 1.4.8, 1.4.16, and 1.4.21, 4.2.3.4, 4.3.2 not related to OTSG replacement.

ANO-1 FSAR change pages were prepared and included in the LBD Change for ER-ANO-2002-1381-000.

The Emergency Plan was reviewed and no changes were identified due to the operation of the ROTSGs. The Quality Assurance Program Manual (QAPM) was reviewed and no changes were identified due to the operation of the ROTSGs [Refs. 1, 5]. An evaluation was completed to assess the impact of the ROTSG installation on the ANO-1 Fire Protection Program. The results of this evaluation indicated that no changes due to the ROTSGs are required for the Fire Hazards Analysis (FHA). All of the TS Bases were manually reviewed for impact due to the ROTSGs [Ref. 4]. No changes to the TS Bases specifically due to the ROTSG are required with the exception of an updated reference to account for a new Overpressure Protection report prepared for the ROTSGs. However, all limits, capacities, and setpoints defined within the TS Bases remain valid and applicable. An evaluation of the ANO-1 SERs results in no updates required for the ROTSGs. The Core Operating Limits Report (COLR) will be revised to reflect the ROTSGs as part of the Cycle 20 core reload analysis and is outside the scope of this review. A review of the Technical Requirements Manual (TRM) and its Bases for ROTSG impact was performed and it was determined that no changes to the current TRM or its Bases are required for ROTSG operation [Ref. 2]. One change was identified but was not related to SG replacement. It is being addressed in TSRC-1-04-03.

A review of NRC SERs (Ref. 1) did not identify any needed changes.

<u>Technical Requirements Manual (TRM)</u>

Section 3.4.3 and associated basis has been updated to include information for the heatup and cooldown during low temperature conditions. A curve for controlling SG pressure for a given secondary temperature was added in accordance with the Appendix G analysis for the ROTSG. [Ref. 5]

<u>Test or Experiments Consideration</u>

ER-ANO-2002-1381-000 does not perform any field installation work. The actual field installation work and associated post modification testing is performed as a part of the installation ERs.

The ROTSGs are very similar to the original OTSGs; thus, the ROTSG is a form, fit, and function match to the original OTSGs. After installation, Entergy Operations will perform certain functional tests on the ROTSGs to verify that they are capable of satisfactorily performing their intended function. [Ref. 76] The following parameters will be verified by functional performance testing:

Thermal and Hydraulic Performance

Each replacement steam generator will meet or exceed the steam temperature, operating level, and stability requirements when supplied with reactor coolant and feedwater as specified in the thermal and hydraulic requirements of the contract specification. These tests will be conducted using plant installed instrumentation and calculations as part of the post installation testing. [Ref. 76]

Reactor Coolant System Flow Rate and Pressure Drop Measurements

Tube-side pressure drop, including pressure losses through the primary inlet nozzles, will result in a reactor coolant system flow that falls within the range specified in the thermal and hydraulic design requirements of the contract specification. These flows will be validated using installed plant equipment and calculations as part of the post installation testing. [Ref. 76]

Primary-to-Secondarv Leakage

pon initial unit operation of the Replacement Steam Generators and during the first operating cycle, leakage from the primary to the secondary side of the steam generator shall be zero (0) gpm, based on applicable plant measurement procedures [Ref. 76].

Post installation testing of the ROTSGs will not result in SSCs being utilized or controlled in a manner that is outside the reference bounds of the design basis as described in the FSAR or is inconsistent with the analyses or descriptions in the FSAR. [Ref. 76] Therefore, per the guidance provided in NEI 96-07, Rev. 1, the post installation testing of the ROTSGs would not be considered a test or experiment.

4. <u>References</u>

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:

LRS 50.59 Unit 1 - Autonomy

Keywords: "steam generator", “SG*, S/G*, OTSG*, “leak before break”, “LBB”, steam generator w/ 10 tube surveillance, “NEI 97-06”, “inconel”, “Alloy 600”, “simulator” , “BAW-1847”, “tube support plates”, “open tube lane”, steam generator w/10 drain, Reg* Guide 1.83, “Inconel”, “Alloy 600”, “Alloy 690”, main feedwater header*, emergency feedwater header*, “downcomer”, “cold leg piping”, “hot leg piping”, “bc”, “bcm”, “boiler condenser”, “”plugging”, “plugging”, “sleeve”, “tube”, “atws”, dose, “efic”, “hydrotest”, “natural circulation”, ”nc”, “rem”, slbic”, “water level”, “afw”, “efw”, “feedwater”, “fw”, “mfw”, “steam”, “emergency feedwater initiation”, “integrated control system”.

LBDs/Documents reviewed manually:

ANO Unit 1 FSAR All Sections, Tables, Figures

ANO Unit 1 Technical Specifications, All Sections
Technical Specification Bases, and
Technical Requirements Manual

5. **Is the validity of this Review dependent on any other change?** (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☐ **Yes**
☒ **No**

If “YES,” list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered “yes,” a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility’s security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility’s telephone or security radio systems?

Documentation for accepting any “yes” statement for these reviews will be attached to this 50.59 Review or referenced below.

D INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation <u>ONLY</u>? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.	☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?	☐ Yes ☒ No

BASIS:

All design basis accidents were evaluated to identify any accidents or transients for which the probability of occurrence could be affected by use of the ROTSGs. A Steam Generator Tube failure event is directly related to the ROTSGs. The probability of occurrence of other accidents could be effected in a more indirect way. The systems external to the ROTSGs that are directly impacted by ROTSG operation are the primary system and the secondary system. Each of these are evaluated for impacts from ROTSG operation separately below:

ROTSGs: All design parameters for the primary side of the steam generator are identical between the ROTSGs and the OTSGs. The secondary side design pressure and temperature for the ROTSG, however, were intentionally increased to 1150 psig (+100 psi) and 605°F (+5°F) to allow for the potential for future operational changes. Relative to the normal operating conditions, assuming the same level of steam generator tube plugging, the normal operating conditions on the primary side are nearly identical. On the secondary side, the overall pressure drop across the OTSG was preserved for the ROTSG design.

The SG tube failure event is caused by a failure internal to the SG. The frequency of occurrence of this accident is governed by tube material, tube dimensions, tube bundle configuration, tube support design, and operating conditions. The increase in the number of SG tubes does not increase the likelihood of a tube failure as no one tube is more or less susceptible to degradation mechanisms than any other tube. The tube dimensions are identical between the OTSG and the ROTSG. The tubes are designed to withstand all operating and design basis accident loads without failure. The ROTSG tubes are designed in accordance with ASME code, made of stronger (higher minimum yield strength), more corrosion-resistant material (Alloy 690), supported as well or better than the OTSG tubes, and joined to the tubesheet by both welding and full-depth expansion. The tube-to-tubesheet joint was qualified to ASME Section III requirements [Ref. 65]. The use of stainless steel tube supports in the ROTSGs minimizes tube denting. Vibration analyses have confirmed that vibration fatigue or wear is adequately addressed in the ROTSG design and is of no concern. For the reasons given above, an ROTSG tube is considered less likely to fail than an OTSG tube. See the description of proposed change section of this review for additional details and comparisions between the operation with ROTSG and the OTSG. Therefore, the frequency of occurrence of a SG tube failure will not increase with the ROTSG operation.

Impacts on Other Portions of the RCS: ER-ANO-2002-1078-015 addresses the cutting of the RCS piping at the OTSGs and re-welding of the piping at the ROTSGs and the hot leg replacement section. There are no other physical changes to the RCS piping in the scope of this ER. Refer to ER-ANO-2002-1078-015 for information on qualification, inspection, and testing of the new hot and cold leg welds for the ROTSGs. The reactor coolant system pressure boundary (RCPB), i.e., the RCS, operating pressure and T_{ave} remains unchanged with the ROTSGs. Evaluation of the limiting (peak RCS pressure) overpressure transients has demonstrated that the ROTSGs do not have an adverse impact upon the ability of the pressurizer safety valves to maintain the RCS pressure below system design pressure [Ref. 22]. Revised inputs to the structural analyses for reactor coolant main loop and tributaries were used to model the effects of the ROTSGs (Refs. 66, 67) concluded that the ROTSGs effect upon the stress and fatigue analyses of the piping and supports is minimal and that resultant loads and fatigue usage factors (including LOCA loads) remain within ASME code allowables [Ref. 41]. The ROTSGs will not create any new conditions or failure mechanisms which would increase the frequency of any events identified as loss of coolant accident (LOCA) initiators. See the description of proposed change section of this review for additional details and comparisions between the operation with ROTSG and the OTSG. Therefore, the frequency of leakage, rupture, or other failure is not increased due to operation of the ROTSGs.

Impacts on Secondary Systems: Other than the cutting of the main steam, main feedwater and emergency feedwater piping at the OTSGs and re-welding of the piping at the ROTSGs there are no other physical changes to the secondary piping. Refer to ER-ANO-2002-1078-012 and 013 for information on qualification, inspection, and testing of the main feedwater and steam line welds for the ROTSGs. The ROTSGs will not cause the main steam or feedwater systems to experience higher pressures during normal operation or during transient conditions, since the setpoints of the main steam safety valves are not changing for the ROTSG design. The feedwater system temperature at full power is not changing as a result of the ROTSG. There will be a slight, +4°F, increase in the nominal operating steam temperature because the tube inspection lane was eliminated. The increase in temperature, however, remains within the current design envelope for all of the attached piping, equipment, and systems external to the steam generator. All external equipment and systems operate in the same design envelope for both the OTSGs and the ROTSGs. See the description of proposed change section of this review for additional details and comparisions between the operation with ROTSG and the OTSG. Therefore the probability of a failure of the seconday systems does not increase due to the operation of the ROTSGs.

(OPEN ITEM -ER HOLD POINT 10.1) Information will be revised as necessary through ERCN. For more information on the effects of FIV, see page 7 of this evaluation, section 3 “Potential for New Failure Modes”.

It is concluded that:

1. The ROTSGs do not increase the frequency of occurrence of an SG tube failure event, the only postulated accident internal to the ROTSG
2. The ROTSGs do not increase the frequency of occurrence of failures of primary system equipment external to the ROTSGs
3. The ROTSGs do not increase the frequency of occurrence of failures of secondary system equipment external to the ROTSGs

Therefore, the operation of the ROTSGs does not increase the frequency of occurrence of an accident previously evaluated in the FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The ROTSGs are designed to the same primary pressure and temperature as the OTSGs. The design conforms to the requirements of ASME code Section III, 1989 Edition with no Addenda and has been reconciled to the original design ASME requirements. The secondary-side design pressure and temperature of the ROTSG was intentionally increased by 100 psi and 5°F respectively to allow for the potential of future operational changes. Relative to the normal operating conditions, assuming the same level of SG tube plugging, the normal operating conditions on the primary side are nearly identical. On the secondary side, the overall pressure drop across the OTSG was preserved for the ROTSG design. For the ROTSG design, there will be a slight (+4°F) increase in the nominal operating steam temperature because the tube inspection lane has been eliminated and leakage past the tube bundle reduced. The increase in temperature, however, remains within the current design envelope for all of the attached piping, equipment, and systems external to the SG. The ROTSGs will therefore not result in any increased challenges to safety systems assumed to function in the accident analyses.

The effects of ROTSGs on other RCS components and plant SSCs were evaluated, including a revision of the ASME Overpressure Protection Report to disposition effects of the ROTSG on the RCS. The Babcock and Wilcox supplied RCS components are compatible with the ROTSGs and do not cause any RCS component to be operated outside of the allowable operating range for the component. All other plant systems and components continue to operate within their design limits. Therefore, likelihood of occurrence of a malfunction of other plant SSCs is not increased by the installation of the ROTSGs.

During postulated FSAR accidents, SSCs important to safety would be required to operate under the same conditions, and perform the same design functions, with the ROTSGs as with the OTSGs. With regard to SSCs assumed to mitigate FSAR accidents, an evaluation of the FSAR accidents with the ROTSGs was performed using the existing initiation time delays, operator guidance, and operating characteristics of the equipment; the evaluation indicated that all SSCs continued to be operated within existing FSAR analysis bounds. Consequently, SSCs important to safety are not called upon to operate under more limiting conditions, or to provide greater mitigation capacities, than currently required with the OTSGs. The increased heat transfer capability of the ROTSGs (no SG tubes are plugged) leads to lower primary temperature and pressure responses to transient conditions. Therefore, use of pressurizer relief devices is less likely.

A structural evaluation demonstrates that all prescribed limits are still met. Therefore, operation of SSCs important to safety would perform their design function following a seismic event.

The effect of the ROTSGs on Reactor Building (RB) post-accident environment was evaluated. The evaluations determined that the responses with the ROTSGs were within existing RB design limits and the post-accident RB environment with the ROTSGs remains within the current OTSG-supported bounds of temperature, pressure, radiation, humidity, flood levels, and chemical addition via building spray. Therefore, the ability of SSCs important to safety within the RB to perform their design function is not degraded with the ROTSGs.

In summary, SSCs necessary for ROTSG operation will operate within the existing operating envelope. SSCs important to safety will be required to operate under the same conditions and to perform the same design functions during accident mitigation as with the OTSGs. Thus, use of the ROTSGs does not result in an increase in the likelihood of occurrence of a malfunction of an SSC important to safety.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The installation and operation of the ROTSGs required the re-evaluation of the design basis accidents and transients to determine if the new design features associated with the ROTSGs (such as flow restricting venturis, decrease in primary inventory, increase in secondary inventory, decrease in overall SG mass, etc.) would have an adverse effect upon the consequences of these accidents and transients. The evaluations determined that there is no increase in the consequences of any accident previously evaluated in the FSAR.

All design basis accidents were evaluated relative to the consequences associated with use of the ROTSGs at ANO-1 (Ref. 11). No design basis accidents would result in dose consequences with the ROTSGs that did not have dose consequences with the OTSGs. The events that have dose consequences that could be affected by use of the ROTSGs are:

Steam Line Break
The SLB resultant doses to the public are caused by the blowdown of the affected steam generator. The dose consequences of a steam line break are a function of the following parameters:

- Primary-to-secondary leak rate.
- Primary coolant activity.
- Iodine activity in the secondary side liquid.
- Mass of secondary steam and feedwater released directly to the atmosphere from the affected steam generator as it blows down.
- Liquid-to-vapor iodine partition coefficient in the secondary side of the affected steam generators.
- Mass of secondary steam released to the atmosphere from the unaffected steam generators to cool the plant.

The primary to secondary leak rate assumed in the dose calculations for ANO-1 is 1 gpm with primary activity assumed as 1 percent failed fuel. The ROTSGs will not affect this assumption. The initial mass inventory of the affected ROTSG is 61,748 lb_m (Ref. 32), while the initial mass inventory of the OTSG is 56,621 lb_m (Ref. 32), giving a total increase in steam generator mass of 5,127 lb_m. The doses listed in the FSAR, however, were determined assuming a conservative initial steam generator inventory of 62,600 lb_m, which bounds both the OTSG and the ROTSG initial secondary-side mass inventory. Therefore, the ROTSG does not affect steam generator secondary mass inventory assumption in the FSAR SLB dose calculations. The iodine activity is limited to 0.17 µCi/gram in the secondary side as defined by LCO 3.7.4 of the Technical Specifications. Since the ROTSG secondary mass inventory has increased over the

OTSG, the maximum iodine activity in the ROTSG could be larger than the OTSG. However, this iodine activity was determined using a steam generator secondary inventory of 62,600 lbm, which is conservatively larger than the initial mass used in both the OTSG or the ROTSG SLB analyses. Long-term steam relief via the turbine bypass system (i.e., cooldown to cold shutdown) will be less for the ROTSG as compared with the OTSG because the ROTSG has significantly less metal that needs to cool down to cold shutdown.

In conclusion, the effects of the increase in secondary inventory at full power for the ROTSG will not adversely affect the SLB dose consequences because the original calculations were performed conservatively and will continue to bound ANO-1 after installation of the ROTSGs.

Steam Generator Tube Failure
The steam generator tube failure or steam generator tube rupture (SGTR) event assumes that a double-ended rupture of one steam generator tube occurs with unrestricted discharge from each end. The plant operating conditions, setpoints, and ECCS flows remain unchanged for this event for the ROTSGs. While the ROTSG tubes have the same inner diameter as the OTSG tubes over the heated length of the tubes, the ROTSG tubes are hydraulically expanded over the entire length of the tubesheet. As the mass loss to the break is directly related to the break area, the leak flowrates from the SGTR event were evaluated to ensure the initial tube flow remains less than the FSAR value of 435 gpm. The reconstruction analysis demonstrated that the SGTR ruptured tube flow as described in the FSAR remains applicable for the ROTSGs.

The majority of the reactor coolant that leaks as a result of the tube failure is condensed in the condenser. The fission products escaping from the main steam safety valves and the condensate are released to the atmosphere. The main fission products released during this accident include iodine, xenon, and krypton, creating a thyroid and a whole body dose hazard. The dose consequences of this accident are a function of:

- Reactor coolant activity
- Leakage from ruptured tube
- Iodine partition factor
- Unaffected steam generator leakage
- Initial secondary activity
- Steam released from unaffected steam generator

Similar to the SLB discussed above, the SGTR FSAR dose analysis was performed assuming 1% failed fuel. The liquid-to-vapor iodine partition factor is based on the pressure, temperature, and pH of the secondary coolant. This is a physical property of iodine in steam/water mixtures and is not affected by the ROTSGs. The normal primary-tosecondary leak rate assumed for the SGTR event reactor dose calculations during normal plant operations is 1 gpm and is not affected by the ROTSGs.

The mass of steam released by the unaffected steam generator through the MSSVs is the same for both the OTSG and the ROTSG. During an SGTR event, steam is relieved through the MSSVs and turbine bypass system to remove primary system sensible heat, fuel sensible heat, and residual core fission power following reactor trip to bring the unit to hot shutdown conditions. The boiling lengths in the steam generators adjust automatically to accommodate this requirement. Because the ROTSGs have no effect on the heat sources, there is no difference in the amount of steam released through the MSSVs. Once the heat load from the primary system drops below the steaming capacity of the turbine bypass system, the MSSVs close. The larger ROTSG secondary side mass inventory has no significant effect on the steam relief through the MSSVs because the additional mass is in the downcomer region and has no effect on heat transfer or steam production rates. The flow-limiting venturis on the ROTSG do not affect the MSSV steaming rate because the maximum steam flow during the event is much less than the critical flow of the venturis. Long-term steam relief via the turbine bypass system (i.e., cooldown to cold shutdown) will be less for the ROTSG as compared with the OTSG because the ROTSG has significantly less metal that needs to cool down to cold shutdown.

Therefore, use of the ROTSGs at ANO-1 does not result in an increase in the consequences of any accident reported in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The evaluations described in the Nuclear Discipline sections above and detailed in Reference 11 demonstrate that the acceptance criteria for the accidents and transients evaluated in the FSAR are met with the ROTSGs. The updated FSAR accidents include the effects of malfunctions of equipment important to safety as required. Use of the ROTSGs does not result in an increase in the dose consequences of any FSAR accident. Therefore, use of the ROTSGs will not increase the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

FSAR accidents can be grouped into two categories: (1) those initiated by failures external to the SGs and (2) those initiated by failures in SG components.

The ROTSGs are attached to the existing plant systems at specific interface points, namely, the primary side nozzles, the feedwater nozzles, the steam nozzles, the drain nozzles, and the level measurement taps. The function of the ROTSGs and attached systems, and the connections of the ROTSGs to the existing systems, including their physical supports, are equivalent to those of the OTSGs. Therefore, use of the ROTSGs at ANO-1 does not create the possibility of a different type of externally-initiated accident.

The design of the ROTSGs conforms to the requirements of ASME code Section III, 1989 Edition with no Addenda and has been reconciled against the OTSG design ASME requirements. The main and emergency feedwater nozzles are designed and evaluated to account for thermal stratification and water hammer concerns. Therefore, failures deemed incredible for OTSG components remain so for the ROTSGs.

The ROTSG tubes are designed in accordance with the ASME code, supported as well or better than the OTSG tubes, and joined to the tubesheet by both welding and full-depth expansion. The use of Alloy 690 tubes in the ROTSGs is expected to greatly decrease tube degradation mechanisms. The tube-to-tubesheet joint has been analyzed and demonstrated acceptable for operating conditions as described in the FSAR. The use of stainless steel tube supports in the ROTSGs minimizes tube denting. Vibration analyses have confirmed that vibration fatigue or wear is adequately addressed in the ROTSG design and is of no concern. Additionally, analyses of potentially degraded tubing has been performed in accordance with NEI 97-06, and allowable defects are shown to meet the acceptance criteria and the plugging limit is consistent with the OTSG plugging criteria specified in the Tech Specs. Given these considerations, no new tube failure mechanism is expected.

(OPEN ITEM -ER HOLD POINT 10.1) Information will be revised as necessary through ERCN. **(OPEN ITEM - ER Hold Point 10.2 for completion of NEI 97-06 analyses)** For more information on these issues, see section 3 of this evaluation, “Potential for New Failure Modes”.

The only accident that can be credibly initiated by a failure internal to the ROTSG is a SG tube failure. This accident is already evaluated in the FSAR for a single tube failure. The probability of occurrence of multiple tube failures is sufficiently low such that they are considered incredible. However, existing operator guidance is provided to mitigate multiple tube failures in both SGs.

In summary, the function of the ROTSGs and attached systems, and the connections of the ROTSGs to the existing systems, is equivalent to those of the OTSGs. Therefore, the ROTSGs at ANO-1 do not create the possibility of an accident of a different type. The only accident that can be initiated by a failure internal to the ROTSG is a SGTR. This accident is already evaluated in the FSAR. Therefore, use of the ROTSGs does not create the possibility of an accident of a different type from any previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

Malfunctions of equipment important to safety previously evaluated in the FSAR could be introduced if the use of the ROTSGs results in physical and/or functional differences in this equipment. The ROTSGs are functionally equivalent to the OTSGs. The minor physical differences between the SG designs do not cause any plant SSCs external to the ROTSGs to operate outside of their current design envelope. The ROTSGs are designed to the same primary pressure and temperature as the OTSGs. The secondary side design pressure and temperature of the ROTSG was intentionally increased by 100 psi and 5°F, respectively. However, the ROTSGs will operate within the original SG design limits (e.g. pressures, temperatures, steam quality, primary system flow, feedwater flow), thus, there is no adverse impact on any SSC. There are no new SSCs required to operate the ROTSGs or mitigate the transient response of the ROTSGs. No plant equipment is required to operate outside the design condition or is required to provide enhanced mitigation (i.e. improved performance characteristics) of postulated accidents. The pressure and temperature responses to postulated pipe ruptures with the ROTSGs are bounded by the conditions used to qualify the existing containment design basis as well as the equipment inside containment. Consequently, the ROTSGs do not affect the ability of an SSC to perform the post-accident design function.

The evaluation of normal and emergency operating procedures demonstrates that no change to operator actions or accident mitigation strategies is required due to the ROTSGs. Therefore, use of the ROTSGs is not anticipated to result in operator actions that could cause a malfunction with a different result than any previously evaluated in the FSAR.

Therefore, it is concluded that use of the ROTSGs at ANO-1 does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

The fission product barriers as defined in the FSAR include the fuel cladding, the reactor coolant system pressure boundary, and the containment, i.e., the RB. A review of the design basis limits of these fission product barriers identified the following limits to be addressed.

Fuel Cladding
There are no physical modifications to the fuel assemblies associated with the installation and operation of the ROTSGs. The normal operating conditions within the reactor vessel remain essentially unchanged with the ROTSGs. Evaluations and reanalyses of the accidents presented in Chapter 14 of the FSAR demonstrated that all events having a minimum DNBR acceptance criterion still meet that acceptance criterion with the ROTSGs. It is expected that zinc addition will mitigate any increased risk for AOA (Axial Offset Anomoly) due to nickel passivation of the ROTSG tubes.

Reactor Coolant Pressure Boundary
The ROTSGs have been designed and analyzed to conform to ASME Code, Section III, 1989 Edition with no Addenda and reconciled against the OTSG design code requirements. The ROTSG tube-to-tubesheet joint has been analyzed and proved acceptable for operating conditions as described in the FSAR. Evaluations and reanalyses of the accidents presented in Chapter 14 of the FSAR demonstrated that all events having a maximum primary pressure acceptance criterion still meet that acceptance criterion with the ROTSGs. There are no additional challenges to the RCPB safety devices due to the ROTSGs. Similarly, all RCPB safety device setpoints and capabilities remain acceptable for use with the ROTSGs for all events as described in the FSAR. Piping stress evaluations demonstrated that loads acting on the RCPB piping, components, and supports do not increase beyond the design basis loads when the ROTSG is introduced into the RCS, or that the stresses present after the ROTSG is introduced into the RCS continue to meet the allowable stresses dictated by the applicable design codes.

Reactor Building
The most significant challenges to the RB design pressure are the *Main Steam Line Failure* and large break *Loss of Coolant Accident* (LOCA). The evaluation of the LOCA event determined that the equivalency between the ROTSG and the OTSG with respect to heat transfer capability and primary side mass inventory and the smaller stored energy of the ROTSG due to less metal mass results in a less severe containment response for the ROTSGs. Reanalysis of the *Steam Line Failure* resulted in less integrated mass and energy release to the containment for the ROTSGs than the OTSGs as currently described in the FSAR. Therefore, there are no RB design basis limits exceeded due to the ROTSGs.

Therefore installation of the ROTSGs at ANO Unit 1 does not result in any design basis limit for a fission product barriers being exceeded or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

RELAP5/MOD2-B&W is an advanced system analysis computer code designed to analyze LOCA and non-LOCA events in Pressurized Water Reactors (PWR). The RELAP5/MOD2-B&W code has been reviewed and approved by the NRC for use on B&W plants with once-through steam generators. (References: BAW-10192P-A, Revision 00, "BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants", and BAW-I0I93P-A, Revision 00, "RELAP5/MOD2-B&W For Safety Analysis of B&W-Designed Pressurized Water Reactors"). ANO-1 is a B&W-designed PWR and the ROTSGs are once-through steam generators. Therefore, the use of RELAP/MOD2 is acceptable.

A structural evaluation of the ROTSG was performed by AREVA Calculation 32-5017980, “ANO-1 SGR RCS Structural Evaluation” (CALC-021381E101-68 (Ref. 46)). This analysis utilized methodologies that are not currently described in the ANO-1 FSAR. However, this methodology was reviewed and approved by the NRC on September 6, 2001 for a similar application (Oconee). In their Safety Evaluation Report, the NRC noted that the approach and methodology used for the reactor coolant loop re-analysis in support of the replacement of the steam generators was reasonable and, therefore, acceptable. One basis of this conclusion was that Duke demonstrate that the leak-beforebreak application is valid for the extended period of operation (20 year license extension). This methodology is
acceptable for use on ANO-1 without further NRC review based on the 10 CFR 50.59 implementation guidance contained in NEI 96-07 based on the completion of a stress analysis for the main coolant loop piping as noted above and on a leak-before-break evaluation. All main coolant loop piping, nozzles, and supports remain qualified in their existing condition for operation with the ROTSGs. An evaluation of the impact of the ROTSGs on the current leak-before-break licensing basis contained in B&W Owners Group Topical Report BAW-1847, Rev. 1 was performed in ER-ANO-2002-1381-008, “ANO-1 ROTSG Piping ER”, for the postulated flaw growth and for the new cold leg elbow extensions. CALC-021381E101-212 evaluated the new hot leg elbows and the new welds in the RCS piping that are installed by ER-ANO-2002-1078-015. Both evaluations determined that BAW-1847, Rev. 1 remained bounding.

Leak-Before-Break
The NRC’s approval of leak-before-break was previously used to reduce the LOCA loadings on fuel assemblies (FSAR Section 3.3.3.3.2.1), to justify pipe whip restraints no longer being required on RCS piping (FSAR Section 4.2.6.6), and for the removal of the wire tie ropes on the reactor coolant pumps (FSAR Section 4.1.2.3). The basis for the NRC approval was the B&W Owners Group Topical Report BAW-1847, Rev. 1. The new material (SA-508 Class 3a forging) that is introduced into the RCS by this ER was evaluated by ER-ANO-2002-1381-008 to verify that the B&WOG report remained bounding. The leak-before-break re-evaluation did not involve new or different methodologies than were described in the FSAR. No other evaluation methodologies that are described in the FSAR are revised or replaced by this modification.

As noted in the Description of Proposed Change, LBB was credited to eliminate dynamic loads from a hot leg or cold leg break on the steam generator upper lateral restraints and lower base support. LBB was credited here consistent with BAW-1847, Rev. 1, so no further approval is required.

Therefore, there is no departure from a method of evaluation described in the FSAR used in establishing the design basis or in the safety or structural analysis.

If any of the above questions is checked “YES,” obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-031

I. OVERVIEW / SIGNATURES

Facility: ANO - Unit 1

Document Reviewed: ER-ANO-2005-0566-002 **Change/Rev.:** 0

System Designator(s)/Description:

MCCW / Main Chiller Cooling Water
FS / Fire Water System
ACW / Auxiliary Cooling Water
DW / Domestic Water

Description of Proposed Activity:

Provide alternate source of makeup to the MCCW Cooling Tower while ACW and Raw (Domestic) Water (normal backup source) are out-of-service. [Alternate source of makeup to the MCCW Cooling Tower to be provided from the Fire Water System].

Check the applicable review(s): **(Only the sections indicated must be included in the Review.)**

☐	EDITORIAL CHANGE of a Licensing Basis Document	Section I
☐	SCREENING	Sections I and II required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, and III required
☒	50.59 EVALUATION (#: 05-031)	Sections I, II, and IV required

Preparer: David E. Torgerson / **ORIGINAL SIGNED BY DAVID TORGERSON** / EOI / EFIN /10-04-05
Name (print) / Signature / Company / Department / Date

Reviewer: John Jehlen / **ORIGINAL SIGNED BY JOHN JEHLEN** / EOI / SYE / 10-04-05
Name (print) / Signature / Company / Department / Date

OSRC: J.R. Eichenberger / **ORIGINAL SIGNED BY J.R. EICHENBERGER** / 10-04-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See LI-101 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	Temporary change to FSAR Figure 9-16 as discussed in Evaluation
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the LBD change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113.			

[1] If "YES," see LI-101. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 evaluation.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ **Yes** ☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113, if applicable. If obtaining NRC approval, document the change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.

3. **Basis**

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

Operating License/Technical Specs

The alternate makeup water source (firewater) proposed for the MCCW Cooling Tower is functionally equivalent to the original water source and will result in no change in how these devices operate. The source of makeup water for the MCCW Cooling tower is below the level of detail described in the OL/TS; therefore, implementation of ER-ANO-2005-0566-002 will not make any statements in the OL/TS inaccurate or untrue.

ER-ANO-2005-0566-002 supports a Temporary Alteration to provide an alternate source of makeup (firewater) to the MCCW Cooling Tower while ACW (Auxiliary Cooling Water) and the Raw Water (Domestic Water) source are both unavailable to provide makeup water to the MCCW cooling Tower. An appropriately rated hose will be utilized (connected) from the FS System piping (Fire Water System) to the MCCW (Main Chiller Cooling Water) System piping. The temporary hose will be connected at hose reel HR-36 or directly to FS-236 Valve (HR-36 may be removed and the hose connected directly to FS-236 if HR-36 is leaking) and routed to MCCW piping at the MCCW Tower MU Filter inlet. The temporary fire pump will provide the required flow capacity for the additional Fire Water system load. The flow demanded for MCCW make up (approximately 150 gpm) is well below the capability of the temporary fire pump. If the permanent fire water pump(s) (P-6A or P-6B) are required to supply water for this temporary alteration, then the pump being utilized is conservatively considered inoperable and the appropriate Fire Spec will be entered (reference OP-1000.152 and Unit 1 FSAR Section 9D.2). In the event of a fire that places a demand on the Fire Water System then actions are to be taken to isolate the temporary alteration from the Fire Water System.

The ACW System, a Service Water subsystem, supplies non-nuclear related cooling water requirements and is isolated in the event of an ES actuation. The ACW System is a non-Q, non Safety Related system that is not utilized in any accident mitigation scenario. Raw Water (Domestic Water) is also a non-Q, non Safety Related system that is not utilized in any accident mitigation scenario. Hose reel HR-36 and valve FS-236 are located on the West wall of the old start-up boiler room (Fire Zone 75-AA), and do not provide protection to any safety related equipment. During the temporary alteration, HR-36 will be unavailable if there is a fire. However, there are two other hose reels in the area (HR-20 and HR-22), and portable ABC/CO2 extinguishers are provided. The fire duration in this zone is low. The combustibles consist of flame resistant cable insulation, jacketing material and transients. This zone is normally unoccupied except for inspections, shift tours and maintenance activities. This zone is equipped with detection and partial suppression (wet pipe sprinkler system over the fuel oil day tank).

FSAR Considerations

The MCCW Cooling Tower makeup system and the Fire Protection system are unaffected by this temporary change with the utilization of the temporary fire pump installation. Therefore, this temporary alteration will not require a change to any LBDs. However, FSAR Figure 9-16 will be made inaccurate during the time the temporary alteration is installed. A change to FSAR Figure 9-16 is not required because of the temporary nature of the alteration.

Tests or Experiments Considerations

The purpose of temporary alteration ER-ANO-2005-0566-002 is to provide a functionally equivalent makeup water source to the MCCW Cooling Tower. ER-ANO-2005-0566-002 does not describe any tests or experiments beyond functional post maintenance testing (PMT) to verify that makeup water has been properly provided to the affected equipment. It is therefore concluded that the proposed change does not involve a test or experiment not described in the FSAR.

4. **References**

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

Electronic search method used:	Keywords:
LRS Common 50.59	(“Fire water”), (“raw water”), (“domestic water”), (MCCW), (chilled water), (m219), (m222), (“hose reel”)

LBDs reviewed manually:

Unit 1 FSAR Fig. 9-16 & 9-9

Unit 1 FSAR Sections 9.8, 9D.2, 9D.5

QAPM, FHA

5. **Is the validity of this Review dependent on any other change?**

☐ **Yes**

☒ **No**

If “YES,” list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any new or different chemicals that are currently not authorized for use by the state regulatory agency?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117 for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following question if one of questions C.1 through C.10 above was answered "yes."

Is a change to the Security Plan required? ☐ **Yes** ☐ **No**

Attach to this 50.59 Review or reference below documentation for accepting a "yes" answer for any of Questions C.1 through C.10, above.

Name of Security Plan reviewer (print / Signature / Data

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING
(NOTE: This section is not applicable to Grand Gulf or Waterford 3 and may be removed from 50.59 Reviews performed for Waterford 3 proposed activities.)

If any of the following questions is answered "YES," a 72.48 Review must be performed in accordance with NMM Procedure LI-112 and attached to this 50.59 Review.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the ISFSI including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power that could potentially impact cask loading or storage activities?
9.	☐	☒	Involve a change to the Fuel Building ventilation that could potentially impact cask loading or storage activities?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Involve fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components power that could potentially impact cask loading or storage activities?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The ACW, Domestic Water, and MCCW systems are not initiators of accidents analyzed in the FSAR. The fire water system is utilized to protect equipment. However, the use of the fire water system to supply make up water to the MCCW cooling tower, will not degrade the fire water system's ability to perform its function due to the small demand required. For conservatism, the fire pump will be considered inoperable if the temporary fire pump is not in use and the temporary connections will be isolated in the event of a fire requiring the use of the fire water system. Therefore, the frequency of occurrence of an accident previously evaluated in the SAR will not be increased by this temporary change.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   Although HR-36 will be unavailable during this temporary alteration, the ability of the fire water system to perform its design function for safety related systems is not degraded because this hose reel is used to mitigate fires in a non-safety related portion of the plant. This area is also protected by HR-20, HR-22 and portable ABC/CO2 extinguishers. Additionally, conservative actions are specified to enhance the system operation in the event of the temporary fire pump out of service or during the required use of the fire system. Therefore, the implementation of this temporary alteration will not increase the likelihood of an occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the SAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   There are no safety related components in the area of the temporary fire hoses. As stated above, the ability of the fire system to perform its function is not degraded due to redundant hose reels in the area. Also, valve FS-236 is a manually actuated valve, and will be isolated in the event of a fire or loss of the temporary fire pump, thereby restoring the fire system to full potential. Therefore, the consequences of an accident previously evaluated in the SAR are not affected.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The ability of the fire water system to perform its design function for safety systems is not degraded. There are no safety related components in the area of the temporary hoses, and there are two hose reels in the area which are available for use (HR-20 and HR-22). Additionally, conservative actions are specified to enhance the system operation in the event of the temporary fire pump out of service or during the required use of the fire system. Therefore, the implementation of this temporary alteration will not increase the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the SAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   This temporary alteration places a fire hose in operation within a non-safety related portion of the plant (Fire Zone 75-AA). This fire zone has been designed with four inch floor drains to remove water in the event of the actuation of a sprinkler system. This is a seismic class II area of the plant, and is not protected in the event of an earthquake. Thus, there are several sources of water which may exist as a result of an earthquake. This Temporary alteration also takes HR-36 out of service, but there are two other hose reels in the area which are available for use (HR-20 and HR-22). The SAR also states that inadvertent rupture of the fire protection system will not cause loss of operation to structures, systems or components important to safety. The temporary hose and connections are located away from safety structures, systems and components. Therefore, the possibility of an accident of a different type than previously evaluated in the SAR will not be created by this change.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The SAR evaluates the fire protection system for line breaks, miss-operation and mitigation of fires which could have an effect on equipment important to safety. The ability of the fire water system to perform its design function for safety structures, systems and components is not degraded or altered. There are no safety related components in the area of the temporary hoses. Additionally, conservative actions are specified to enhance the system operation in the event of the temporary fire pump out of service or during the required use of the fire system. Therefore, the implementation of this temporary alteration will not create the possibility for a malfunction of a structure, system, or component important to safety with a different result previously evaluated in the SAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

   BASIS:

   The temporary alteration will not impact any fission product barriers in any way. The fire water system will remain operable. Therefore, the design basis limit for a fission product barrier as described in the SAR can not be exceeded or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

   BASIS:

   The temporary alteration implementation will have no effect on any method of evaluation for any design basis established in the SAR.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2005-032

I. OVERVIEW / SIGNATURES

Facility: ANO

Document Reviewed: ER-ANO-2004-0020-000, CRDM Replacement **Change/Rev.: 0**

System Designator(s)/Description: Control Rod Drive Mechanisms

Description of Proposed Change:

The components being replaced by ER-ANO-2004-0020-000 include the control rod drive mechanisms (CRDM's), the control rod position indicators (PIs) and the PI amplifier module assemblies.

The CRDMs to be installed are designated as a "Type C" drive by the manufacturer. Three Type C drives are currently installed and the remaining 65 are "Type B". Design feature differences between the Type B and Type C drives include a change in material and wall thickness of the center section of the motor tube and design of the stator. The Type B drives utilized an inconel clad carbon steel motor tube center section, while Type C drives utilize martensitic stainless steel. The Type C CRDM stator design was updated in conjunction with the change in motor tube center section material. Other design features changes of note from Type B to Type C include rotor assembly modifications to simplify the segment arm, motor tube base configuration (no change to interface with CRDM flange) and a 0.75" increase in overall length. The increase in length results in a slightly longer reach for control rod coupling/uncoupling evolutions, thus enhancing the compatibility with alternative fuel assembly designs.

This change affects the CRDM absolute PI devices. Original plant design included "Type B" PIs (not to be confused with Type B CRDMs). A significant number (49 of 68) of the original PIs have previously been replaced with a newer design employing redundant reed switch strings This newer design is designated by the manufacturer as "Type R4C". This change will result in Type R4C PIs being employed on all 68 CRDMs.

A change from a Type B PI to a Type R4C PI requires replacement of the associated amplifier module assembly. The replacement amplifier module assembly is designed to allow utilization of the redundancy available in the R4C PI (i.e., can be switched to single string output from the normal 2-string combined mode output). Because these amplifier module assemblies have been previously installed in conjunction with R4C PI installations, only 19 of 68 Type B amplifier module assemblies require replacement with Type R4C as a result of this change.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-032)**	**Sections I, II, and IV required**

Preparer: Randall S. Smith / **ORIGINAL SIGNED BY RANDALL SMITH** / Univ. Personnel/RVCH / 11-09-05
Name (print) / Signature / Company / Department / Date

Reviewer: Jerry W. Howell / **ORIGINAL SIGNED BY JERRY HOWELL** / EOI / SG-RVCH / 11-09-05
Name (print) / Signature / Company / Department / Date

OSRC: J. R. Eichenberger / **ORIGINAL SIGNED BY J. R. EICHENBERGER** / 11-10-05
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings and 50.59 Evaluations.]

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM ENS-LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	***3.2.4.3.2.A/J, fig 3-70, 4.3.2, 7.2.2.2.1, 7.2.2.3.1, 7.2.2.3.4, Figure 7-21***
TS Bases	☒	☐	***B.3.1.4***
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES", perform an Exemption Review per Section III OR perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM ENS-LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM ENS-LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. **Does the proposed activity involve a test or experiment not described in the FSAR?**

☐ **Yes**
☒ **No**

If "yes," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. **Basis**

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

Operating License/Technical Specifications

The control rod drive system, in particular control rod drop times and rod position limits, are addressed by the OL/TS.

The control rod drop time limit of 1.66 seconds per SR 3.1.4.3 is not affected. There are differences in the design criteria for drop time values for a Type C CRDM vs a Type B CRDM, however the Type C can and will continue to meet the less than or equal to 1.66 seconds criteria. In practice, the currently installed Type C CRDMs have drop times that fall within the range of Type B CRDMs drop times.

FSAR

Sections 3, 4 and 7 of the FSAR are affected by the proposed change. The changes required relate to CRDM motor tube material, descriptive text encompassing Type B and Type R4C position indicator designs and figure changes.

Tests or Experiments Considerations

The proposed change does not involve a test or experiment. All of the replacement components have previously been employed and are encompassed by existing plant procedures.

4. **References**

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.5.1[5](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LRS 50.59 Unit 1	"buffer amplifier","amplifier module","motor tube", "leadscrew","bayonet coupling","stator temperature", "monofilar","bifilar","drop time","R4C","position indicator", "position indication","Type B","Type C", "CRDM*","control rod drive","crd","403","martensitic", "alloy 82","insertion time","time to 2/3 insertion"

LBDs/Documents reviewed manually:

FSAR Sections 3.2, 4.3, 7.2, 14

5. Is the validity of this Review dependent on any other change?

☐ **Yes**

☒ **No**

If "YES", list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure ENS-EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure ENS-EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

(NOTE: This section is not applicable to Waterford 3 and may be removed from 50.59 Reviews performed for Waterford 3 proposed activities.)

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure ENS-LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: _____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions.

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

There are six (6) accidents described in Chapter 14 of the FSAR which bear consideration for potential impact with respect to this change. They are the Startup Accident (Section 14.1.2.2), the Rod Withdrawal Accident at Rated Power Operation (Section 14.1.2.3), the Moderator Dilution Accident (Section 14.1.2.4), the Stuck-Out, Stuck-In, or Dropped Control Rod Accident (Section 14.1.2.7), the Rod Ejection Accident (Section 14.2.2.4) and the Loss of Coolant Accident (Section 14.2.2.5). Each is addressed separately as follows;

Startup Accident

CRDMs - The control rod drive control system (CRDCS) is unaffected by CRDM replacement beyond a reduction in CRDM power requirements and as such no new control system failures modes are introduced. The replacement CRDMs are designed to respond to the control system commands in exactly the same manner as those being replaced. Protective device operation resulting in CRDM stop, insertion or trip is unaffected and the replacement CRDMs provide the same response modes as those replaced. There is no impact to the frequency of occurrence of an uncontrolled reactivity addition.

PIs – As a non-safety related device, the CRDM PIs are not relied upon for accident mitigation, however, as an input to the CRDCS the PIs could be construed as potential contributors to an uncontrolled reactivity addition event. The replacement PIs have the same principle of operation, physical characteristics, mounting method and location, etc. as those replaced. Any reduction in accuracy introduces no new failure mode which increases the likelihood of an uncontrolled reactivity addition because the change in accuracy does not introduce any new possibility of uncontrolled or undesirable rod movement.

PI Amplifier Modules – The amplifier modules serve only as an interface device to transmit the PI signal to the CRDCS. The replacement modules have an additional feature to allow bypass of one of the two output strings. Reliability is enhanced through redundancy, however, operation with the output string bypass feature enabled is at a lower accuracy than the Type B PIs replaced. This allowable reduction in accuracy on individual PIs does introduce any new group or system failure modes and does not impact the potential frequency of occurrence of an uncontrolled reactivity addition.

Rod Withdrawal Accident at Rated Power Operation

CRDMs – Aside from a reduced power consumption, the CRDCS is unaffected by CRDM replacement and as such no new control system failures modes are introduced. The replacement CRDMs are designed to respond to the control system commands in exactly the same manner as those being replaced. Protective device operation resulting in CRDM stop, insertion or trip is unaffected and the replacement CRDMs provide the same response mode as those replaced.

PIs – As non-safety related devices, the CRDM PIs are not relied upon for accident mitigation, however, as an input to the CRDCS the PIs are potential contributors to a rod withdrawal accident. The decrease in accuracy with the Type R4C bypass feature enabled must be addressed in this regard. One of the ways in which the possibility of an inadvertent rod withdrawal is minimized is through the sequence interlock section of the CRDCS. The PIs provide the basic input to the sequencer. The group average signal serves as input to electronic setpoint trip units which are activated at approximately 25% and 75% of group rod withdrawal. Outputs of these bistables actuate enable relays which permit the rod groups to be commanded. Since the group average signal has been determined to be within error assumptions of the safety analyses, the function of the sequencer is not considered to be significantly affected. [The supporting documentation for this assertion and other supporting information is included in the 50.59 evaluations for DCP-86-1030, PI Tube Change and DCP-88-1035, Group 8 PI Tube Replacement. These two documents provided the original basis for use of Type R4C PIs including the assertion that the potential frequency of occurrence of a rod withdrawal accident is not increased.]

PI Amplifier Modules – The amplifier modules serve only as an interface device to transmit the PI signal to the CRDCS. Reliability is enhanced through redundancy, however, operation with the output string bypass feature enabled is at a lower accuracy than the Type B PIs replaced. This allowable reduction in accuracy on individual PIs does introduce any new group or system failure modes and does not impact the potential frequency of occurrence of a rod withdrawal accident.

Moderator Dilution Accident

This event is initiated by adding deborated make-up water to the RCS. Therefore, changes to the CRDMs, PIs, and PI Amplifier Modules will have no effect on the probability of the initiation of the MDA. Changes to these components could effect the mitigation of the MDA, however.

Stuck-Out, Stuck-In, or Dropped Control Rod Accident

CRDMs - The control rod drive control system (CRDCS) is unaffected by CRDM replacement and as such no new control system failures modes are introduced. CRDM operation is, in all relevant aspects, the same for the Type B and Type C CRDMs. The replacement CRDMs are designed to respond to the control system commands in exactly the same manner as those being replaced. Protective device operation resulting in CRDM stop, insertion or trip is unaffected and the replacement CRDMs provide the same response as those replaced. The Type B/Type C deltas are characterized as design improvements for the Type C, with no difference in the actual mechanical operation of the CRDM. The differences in the two types are transparent to plant operations with the exception of reduced power consumption for the Type C CRDMs. Type C CRDMs have been in use on a limited basis at ANO and in service for decades at other B&W NSSS plants with no identified trend of stuck or dropped rods occurring at any frequency higher than the Type B CRDMs being replaced.

PIs – As a non-safety related device, the CRDM PIs are not relied upon for protective action. The replacement PIs have the same principle of operation, weight, mounting method and location, etc. as those replaced. Any reduction in accuracy does not introduce any new failure mode which increases the likelihood for a stuck or dropped rod.

PI Amplifier Modules – The amplifier modules serve only as an interface device to transmit the PI signal to the CRDCS, with the Type R4C PI compatible design having the additional feature to allow bypass of one of the two Type R4C PI output strings. Reliability is enhanced through redundancy, however, operation with the output string bypass feature enabled is at a lower accuracy than the Type B PIs replaced. This allowable reduction in accuracy on individual PIs does not introduce any new failure modes and does not impact the potential frequency of occurrence of a rod withdrawal accident.

Rod Ejection Accident

CRDMs – The rod ejection accident is related to a circumferential failure of the CRDM motor tube at any point at or above the connecting flange. The Type C CRDM motor tube uses a different material of construction for the center section as well as slightly different wall thicknesses and lengths. The Type C CRDM motor tube is designed to the same criteria as the existing Type B CRDM motor tube, is qualified to the applicable ASME Code, has been subjected to the same analysis and testing and has significant operating experience at ANO and elsewhere. There is no evidence that any increase in the likelihood of this event is present in conjunction with the use of the Type C CRDM. The use of Type 403 stainless steel as replacement for the clad carbon steel in the motor tube center section is considered an enhancement in this regard due to the elimination of Alloy 600 and associated failure mechanisms in this area.

PIs – There is no relevance for this accident.

PI Amplifier Modules – There is no relevance for this accident.

Loss of Coolant Accident

CRDM – See Rod Ejection Accident pressure boundary integrity discussion above.

PIs – There is no relevance for this accident.

PI Amplifier Modules – There is no relevance for this accident.

Conclusion; This change does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

CRDMs - The CRDMs are in general treated as a safety related piece of equipment, although certain piece parts have previously been determined to not serve a safety related function. In particular, the reactor coolant system pressure boundary aspect and the trip features are safety related. SSC's important to safety that the CRDMs interface with include the reactor vessel closure head CRDM flanges, the service structure seismic tie plates (serve to restrain the upper end of the CRDMs at the top of the service structure), the reactor vessel head high point vent system, the CRDCS and the control rod assemblies. In each instance the changes associated with Type C CRDM do not affect the interfacing equipment with the exception of the lower power consumption from the CRDCS. In no instance is the likelihood of occurrence of malfunction of an SSC important to safety previously evaluated in the FSAR affected.

PIs – The PIs are not safety related and that status is not changed. Therefore PI failure cannot by definition preclude the operation of SSC's important to safety. The PIs are mounted to reactor coolant pressure boundary components (CRDMs) above the reactor vessel head. However, the seismic II/I relationship is unaffected since the mounting method, physical characteristics and mounting location is the same for the Type R4C PIs as for the Type B PIs. There is no basis to conclude that this change could increase the likelihood of the occurrence of a malfunction of equipment important to safety.

PI Amplifier Modules – The PI amplifier modules are not safety related and are part of a non-safety related system (position indication). That status is not changed and therefore, by definition, failure of the PI amplifier module cannot preclude the satisfactory operation of equipment important to safety or contribute to an increased likelihood of malfunction.

Conclusion; This change does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The following accidents described in Chapter 14 of the FSAR were considered for potential impact with respect to this change. They are the Startup Accident (Section 14.1.2.2), the Rod Withdrawal Accident at Rated Power Operation (Section 14.1.2.3), the Moderator Dilution Accident (Section 14.1.2.4), the Stuck-Out, Stuck-In, or Dropped Control Rod Accident, the Rod Ejection Accident (Section 14.2.2.4) and the Loss of Coolant Accident (Section 14.2.2.5). Each is addressed separately as follows;

Startup Accident

CRDMs - There is no change to the rod overlap controls and therefore introduce no associated change in the reactivity addition rate. The control rod withdrawal rates are also unchanged and likewise introduce no change in the reactivity addition rate potential. Also, protective features such as startup rate interlocks, alarms and trips are unaffected by the conversion to a full complement of Type C CRDMs. There is a change in the rod drop time design criteria, with the Type C rod drop time being longer. Consideration of rod drop time differences and the effect on the safety analyses is considered in A1-NE-2004-002-00, ANO-1 Cycle 20 Safety Analysis Groundrules. Current rod drop time acceptance criteria is not affected. The consequences of an uncontrolled reactivity addition remain unaffected for all other aspects of this evaluation.

PIs – As a non-safety related device, the CRDM PIs are not relied upon for protective action. Any reduction in accuracy associated with Type R4C PI operation introduces no new failure mode which increases the consequences of an uncontrolled reactivity addition.

PI Amplifier Modules – The amplifier modules serve only as an interface device to transmit the PI signal to the CRDCS, with the additional feature to allow bypass of one of the two PI output strings. Reliability is enhanced through redundancy, however, operation with the output string bypass feature enabled is at a lower accuracy than the Type B PIs replaced. This allowable reduction in accuracy on individual PIs does introduce any new group or system failure modes and does not impact the potential consequences of an uncontrolled reactivity addition.

Rod Withdrawal Accident at Rated Power Operation

CRDMs – Aside from a reduced power consumption by the CRDMs, the CRDCS is unaffected by CRDM replacement and as such no new control system failures modes are introduced. The replacement CRDMs are designed to respond to the control system commands in exactly the same manner as those being replaced. Protective device operation resulting in CRDM stop, insertion or trip is unaffected and the replacement CRDMs provide the same response as those replaced with no effect on the consequences of a rod withdrawal accident. There is a change in the rod drop time design criteria, with the Type C rod drop time being longer. Consideration of rod drop time differences and the effect on the safety analyses is considered in A1-NE-2004-002-00, ANO-1 Cycle 20 Safety Analysis Groundrules. Current rod drop time acceptance criteria is not affected. The consequences of an uncontrolled reactivity addition remain unaffected for all other aspects of this evaluation.

PIs – As a non-safety related device, the CRDM PIs are not relied upon for protective action, however, as an input to the CRDCS the PIs could be construed as potential contributors to a rod withdrawal accident. The replacement PIs have the same principle of operation, physical characteristics, mounting method and location, etc. as those replaced. Any reduction in accuracy introduces no new failure mode which increases the consequences of the rod withdrawal accident.

PI Amplifier Modules – The amplifier modules serve only as an interface device to transmit the PI signal to the CRDCS. Reliability is enhanced through redundancy, however, operation with the output string bypass feature enabled is at a lower accuracy than the Type B PIs replaced. This allowable reduction in accuracy on individual PIs does introduce any new group or system failure modes and does not impact the consequences of a rod withdrawal accident.

Moderator Dilution Accident

CRDMs - Rod insertion limits will continue to function to isolate the dilution flowpath. With no control rod insertion (e.g., rods in manual with no operator intervention) the reactor will continue to be tripped on high RCS pressure. The control rod drive control system (CRDCS), aside from a reduced power consumption by the CRDMs, is unaffected by CRDM replacement and as such no new control system failures modes are introduced. The replacement CRDMs are designed to respond to the control system commands in exactly the same manner as those being replaced. Protective device operation resulting in CRDM stop, insertion or trip is unaffected and the replacement CRDMs provide the same response as those replaced in mitigating a moderator dilution accident.

PIs – As a non-safety related device, the CRDM PIs are not relied upon for protective action, however a moderator dilution accident is mitigated by control rod insertion. The change to Type R4C PIs does not impact the ability of the control rods to respond and as a result does not affect the response to a moderator dilution accident.

PI Amplifier Modules – The amplifier modules serve only as an interface device to transmit the PI signal to the CRDCS, with the Type R4C having the additional feature to allow bypass of one of the two output strings. The change to Type R4C PI compatible amplifier modules has no affect on the CRDCS response to a moderator dilution accident.

Stuck-Out, Stuck-In, or Dropped Control Rod Accident

CRDMs - The control rod drive control system (CRDCS) is unaffected, other than reduced CRDM power consumption, by CRDM replacement and as such no new control system failures modes are introduced. CRDM operation is, in all relevant aspects, the same for the Type B and Type C CRDMs. The replacement CRDMs are designed to respond to the control system commands in exactly the same manner as those being replaced. Protective device operation resulting in CRDM stop, insertion or trip is unaffected and the replacement CRDMs provide the same response as those replaced. The Type B/Type C deltas are characterized as design improvements for the Type C, with no difference in the actual mechanical operation of the CRDM. The differences in the two types are transparent to plant operations with the exception of reduced power consumption for the Type C CRDMs. Type C CRDMs have been in use on a limited basis at ANO and in service for decades at other B&W NSSS plants with no identified occurrence of stuck or dropped rods at any frequency greater than the Type B CRDMs being replaced. Additionally, the stuck-out, stuck-in or dropped rod accident analysis assumes no Control Rod Drive System or ICS action occurs and as such no mitigation action is assigned to the unaffected CRDMs for the stuck-out, stuck-in or dropped rod accident.

PIs – As a non-safety related device, the CRDM PIs are not relied upon for protective action, however, as an input to the CRDCS the PIs could be construed as potential contributors to a rod withdrawal accident. The replacement PIs have the same principle of operation, physical characteristics, mounting method and location as those replaced. Any reduction in accuracy does not introduce any new failure mode which increases the likelihood for the stuck-out, stuck-in or dropped rod accident.

PI Amplifier Modules – The amplifier modules serve only as an interface device to transmit the PI signal to the CRDCS, with the Type R4C having the additional feature to allow bypass of one of the two output strings. Reliability is enhanced through redundancy, however, operation with the output string bypass feature enabled is at a lower accuracy than the Type B PIs replaced. This allowable reduction in accuracy on individual PIs does not introduce any new group or system failure modes and does not impact the potential frequency of occurrence of a stuck-out, stuck-in or dropped rod accident.

Rod Ejection Accident

CRDMs – The rod ejection accident is related to a circumferential failure of the CRDM motor tube at any point above the connecting flange. The Type C motor tube uses a different material of construction for the center section as well as slightly different wall thicknesses and lengths. The Type C CRDM motor tube is designed to the same pressure boundary criteria as the existing Type B CRDM motor tube and is qualified to the applicable ASME Code. The use of Type 403 stainless steel as replacement for the clad carbon steel in the motor tube center section is considered an enhancement with respect to the elimination of Alloy 600 and the associated potential failure mechanisms in this reactor coolant pressure boundary application. There is a change in the rod drop time due to design differences, with the Type C rod drop time being longer. Consideration of rod drop time differences and the effect on the safety analyses is considered in A1-NE-2004-002-00, ANO-1 Cycle 20 Safety Analysis Groundrules. Current rod drop time acceptance criteria are not affected. The consequences of an uncontrolled reactivity addition remain unaffected for all other aspects of this evaluation.

PIs – The Type R4C PI has the same physical characteristics, the same mounting methods and same mounting locations as the Type B it replaces. There is no recognized mechanism in which the change could potentially affect the consequences of a rod ejection accident.

PI Amplifier Modules – There is no relevance for this accident.

Loss of Coolant Accident

CRDM – See Rod Ejection Accident discussion above. The loss of coolant accident also introduces considerations of other potential failure modes beyond those associated with the rod ejection accident (e.g., motor tube axial rupture or closure assembly failure). The closure assemblies are re-used from the existing assemblies and are attached in the same manner as to the Type B CRDMs. The conclusion remains the same in that there is no evidence that any increase in the likelihood of a loss of coolant accident is present as a result of the use of the Type C CRDM.

PIs – See the Rod Ejection Accident discussion above.

PI Amplifier Modules – There is no relevance for this accident.

Conclusion; This change does not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

CRDMs – There is no change in the assignment of safety functions to the CRDMs in conjunction with this change. Additionally, there is no change in the degree of reliance on the CRDMs to perform their assigned safety functions. Operation is unchanged and the substitution of Type C CRDMs for Type B's is transparent to the interfacing components and systems with the exception of reduced power consumption from the CRDCS. As a result, the consequences (radiological releases) of a malfunction of a structure, system or component important to safety and previously evaluated in the FSAR are not affected.

PIs – The PIs are not safety related components and this change does not affect that classification. Additionally, the change to the Type R4C PI does not change the reliance on any associated equipment, including the CRDCS, with respect to performance of any safety function. While the control rod assembly position indication is considered a regulatory/safety significant function (i.e., Reg Guide 1.97), the effects of the change in accuracy from Type B to Type R4C PIs on system performance have previously been evaluated and found acceptable (Reference DCP-86-1030 and DCP-88-1035) and do not constitute a potential increase in the consequences of malfunction of any SSC important to safety.

PI Amplifier Modules - The PI amplifier modules are not safety related components and this change does not affect that classification. Additionally, the change to the Type R4C PI compatible amplifier modules does not change the reliance on any associated equipment, including the CRDCS, with respect to performance of any safety function. The implication of the differences in accuracy from that seen with the Type B PIs and associated amplifier modules, in particular when employing the output string bypass feature, has been previously evaluated and found acceptable (Reference DCP-86-1030 and DCP-88-1035) with respect to any potential increase in the consequences of a malfunction of an SSC important to safety and previously evaluated in the FSAR.

Conclusion; This change does not result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

No accidents previously considered incredible or otherwise not considered in the FSAR are considered to become more likely as a result of this change. All relevant aspects of the affected CRDM and PI components and their interrelationship with other SSC's are encompassed in the current FSAR accident analysis, including, but not limited to, structural failure of the CRDM pressure boundary, failure of the CRDM to respond to CRDCS commands, dropped rods and failure of the PI system to indicate true rod position (including the effect of reduced PI accuracy).

Conclusion; This change does not create a possibility for an accident of a different type than any previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The interrelationships with equipment important to safety are not affected at a level approaching the threshold of creating a malfunction with a different result. The replacement components themselves do not exhibit or portend any malfunction possibilities with a different result than any previously evaluated. CRDM operation will not change. No malfunctions resulting in conditions outside the existing bounding LOCA and rod ejection analyses, control rod operation anomalies or position indication errors affecting CRDCS operation can be credibly postulated.

Conclusion; This change does not create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?

☐ Yes
☒ No

BASIS:

The design bases limit for the fuel cladding is not affected by this change. The DNB analysis addressed the change in drop times in the Cycle 20 reload report. The resulting parameters remain within the existing safety analyses such that the design bases limit is not violated.

The design bases containment pressure limit is not affected. Likewise, the containment pressure achieved in the event of any design bases accident is not affected and accordingly will not be caused to exceed the design bases limit.

The CRDMs do constitute a portion of the reactor coolant system pressure boundary and as such replacement of the CRDMs could have an impact to the RCS boundary fission product barrier. The pressure boundary component affected is the CRDM motor tube. The closure assembly, a pressure boundary component located at the top of the motor tube is not replaced and is transferred from the existing CRDMs to the replacement CRDMs. The Type C CRDMs have been designed, constructed, tested and purchased to the same criteria as the existing Type B CRDMs. Additionally, a stress report summary has been previously generated to confirm the structural acceptability of the Type C CRDMs for the existing design basis limit. This change does not affect that limit. Likewise, the rod ejection design basis is unaffected (see previous responses).

Conclusion; This change therefore does not result in design basis limit for a fission product barrier as described in the FSAR being exceeded or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☒ No

BASIS:

This change does not affect any method of evaluation and as a result can not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

If any of the above questions is checked "YES", obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure ENS-LI-113.

ANO 50.59 Evaluation Number

2005-033

I. OVERVIEW / SIGNATURES

Facility: ANO-1

Document Reviewed: ER-ANO-2004-0491-000, including CALC-A1-NE-2005-001 ("ANO-1 Cycle 20 Reload Report") & CALC-A1-NE-2005-002 ("ANO-1 Cycle 20 COLR") & CALC-ANO-ER-05-22 ("ANO-1 Mk-B-HTP & Mk-B9 LOCA Summary Report") **Change/Rev.:** 000

System Designator(s)/Description:
REAN

Description of Proposed Activity:
The Cycle 20 Reload Report (CALC-A1-NE-2005-001) is prepared to document changes in nuclear, thermal-hydraulic, and mechanical design of the reactor core. As such, the reload report provides the bases for the startup testing and operation of the Cycle 20 fuel cycle design. It is based on the results of safety analyses performed by Areva. All analyses and assessments were performed using NRC-approved methodologies. This reload marks the beginning of the transition to Mark-B-HTP fuel with M5 cladding. Two reload-driven Technical Specification changes were required to implement the Cycle 20 fuel cycle design. Those changes were to ANO-1 TS 2.1.1.2 to permit the use of the BHTP correlation and TS 4.2.1 to permit the use of M5 advanced alloy for fuel rod cladding and fuel assembly structural components. Use of the BHTP correlation constitutes the only analyses methodology change identified for the next fuel cycle design. The NRC has issued an SER authorizing both Technical Specification changes and use of the BHTP correlation for ANO-1 (Amendment 226). The Loss of Coolant Accident (LOCA) analysis, incorporating Mark-B-HTP fuel and the new OTSG design, has also been performed for Cycle 20 and beyond and supports this Reload Report. Thus, the LOCA Summary Report (ANO-ER-05-22) will be addressed in this 50.59 evaluation. The Cycle 20 Reload Report (RR) will replace the current SAR chapter 3A (Cycle 19 RR) in its entirety; numerous, additional SAR changes have also been identified.

ANO-1 TS 5.6.5 requires that the core operating limits be established for Cycle 20 operation. The core operating limits report (COLR) was prepared using data from the reload report and other applicable references for Cycle 20 design. As allowed by USNRC Generic Letter 88-16, the Cycle 20 COLR (CALC-A1-NE-2005-002) contains certain cycle-specific values such as the core operating limits, protective limits, and trip setpoints. Most limits and setpoints contained in the COLR are changed for Cycle 20 due to the insertion of Mark-B-HTP fuel into the core.

Some Technical Specification Bases changes have also been identified and will be evaluated as part of this 50.59 review. Those changes involve clarification on the use of the BHTP correlation, consistent with the approved Technical Specification revisions, and several reference changes.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-033)**	**Sections I, II, and IV required**

Preparer: Daniel W. Fouts / **ORIGINAL SIGNED BY DAN W. FOUTS** / ENS / ENG / 11-10-05
Name (print) / Signature / Company / Department / Date

Reviewer: Robert W. Clark / **ORIGINAL SIGNED BY ROBERT W. CLARK** / ENS / ENG / 11-12-05
Name (print) / Signature / Company / Department / Date

OSRC: J. R. Eichenberger / **ORIGINAL SIGNED BY J. R. EICHENBERGER** / 11-18-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See LI-101 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	1.5.4, 1.10, 3.1.2.3, 3.2.3.1, 3.2.3.1.1, 3.2.4.2.1, 3.3.3.3.2.1, 3.4, Fig 3-66C(new), Appendix 3A (all), 14.1.2.6.2, 14.2.2.5, 14.4, Figures 14-34, 14-35, & 14-36
TS Bases	☒	☐	B2.1.1, B3.2.5, B3.4.1
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☒	☐	All
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the LBD change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113.			

[1] If "YES," see LI-101. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 evaluation.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☐ **Yes**
☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113, if applicable. If obtaining NRC approval, document the change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.

3. Basis

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

General Reload Information:

The Cycle 20 fuel cycle design has a maximum design length of 482 EFPD, which includes a reactor coolant system (RCS) average temperature reduction maneuver near the end of the cycle. The APSRs will be completely withdrawn from the core prior to the average temperature reduction maneuver. The core design includes the insertion of 56 Areva Mark-B-HTP NR-LEF fresh fuel assemblies (Batch 22). Sixty (60) once-burned Mark-B9ZL NR-LEF assemblies (Batch 21) and 60 twice-burned Mark-B9ZL NR-LEF assemblies (Batch 20) are shuffled to new core locations, with a center Mark-B9ZL NR-LEF assembly from Batch 18 completing the 177 fuel assembly core. No core asymmetries will be introduced based on this core design.

The Reload Report (CALC-A1-NE-2005-001) is prepared to document changes in nuclear, thermal-hydraulic, and mechanical design of the reactor core necessary to implement the Cycle 20 fuel cycle design. As such, the noteworthy changes in reload design and analyses are summarized below:

- The mechanical design of the fuel assemblies is changed from previous cycles of operation (the control components are not). Cycle 20 includes the introduction of 56 Mark-B-HTP fuel assemblies that contain M5 clad and intermediate grid straps. M5 is an improved zirconium alloy which increases corrosion resistance of the fuel rods and provides greater dimensional stability for rod cladding and assembly components. The enrichment levels of the fuel were decreased slightly to meet the design cycle length of 482 EFPD. The total number of BPRAs remained the same (40) with slightly different burnable poison concentrations.

- Most grid straps in the new design are made of two stamped pieces of M5 material that form small flow channels in the straps. Along with strengthening the grid straps, these channels provide some swirling in the coolant flow through the assemblies. The sides of the flow channels provide the support surfaces for the fuel rods in the grids. A thicker side plate is used to make the outside perimeter of the new grid design. The double grid straps and thicker side plate makes a structurally stronger mid-grid than the previous Mark-B9ZL design with more contact area for the fuel rods in the grid. The combination of the stronger grid and more fuel rod contact area provides additional margin to prevent fuel damage from handling and fuel failure from grid fretting.

- The Mark-B-HTP fuel assembly also includes a new debris resistant lower end fitting. The FUELGUARD end fitting has a curved blade flow area that provides a tortuous path for debris to enter the fuel assembly while minimizing the pressure drop. A rigid frame holds the curved blades and provides the structural strength required for the lower end fitting.

- The upper end fitting for the Mark-B-HTP design has been changed from the helical holddown spring design to a six-leaf holddown spring design. The new design retains the ability for reconstitution by removing the upper end fitting. Fuel assembly handling was not changed with the new design. The upper end fitting flow pattern was modified to support the leaf spring footprint. The new grid design and end fitting changes increase the pressure drop across the assembly. This is a necessary trade-off to improve the grid properties and is reflected in similar experience with other vendor-improved grid designs.

- The Guide tubes and Instrument tubes for the Mark-B-HTP fuel assembly are made from M5 material and have been modified as needed to interface with the new upper and lower end fitting. A small drain hole has been added to the lower end plug to aid in coolant removal for dry cask storage.

- The assembly is slightly longer than the current design to reflect the lower growth of the M5 fuel rods and guide tubes with radiation, but is still designed to fit the same envelope as the current assemblies with mid- grids at the same elevation. The new design has a slightly higher pellet density, larger pellet diameter, and longer stack length. These changes add about 25 kg nominally to each fuel assembly.

- The angled flow channel in the mid-grid straps was originally developed as a flow mixing feature to improve assembly thermal-hydraulics. The design is not as effective as other mixing vane designs and Areva is claiming little to no improvement in DNBR margins.

- Insertion of Mark-B-HTP fuel requires adoption of the BHTP correlation for predicting DNB, and the location of DNB. The minimum acceptable value of DNBR for the BHTP correlation is 1.132.

- The slight reduction to the loss of coolant accident (LOCA) linear heat rate target (not operating) limits (LHR) for Batch 21 fuel continues. This reduction is necessary to comply with the limitation on power peaking factors at the limits of normal operation as specified in the SER to BAW-10192P-A. The maximum reduction for the Batch 21 fuel is 0.27 kw/ft. The LOCA linear heat rate limits for Batch 22 fuel requires no reduction. The ANO-1 LOCA Summary Report for the Mk-B-HTP and Mk-B9 fuel demonstrated that the 5 criteria of 10CFR50.46 were met for this reload.

- One assembly from Cycle 19 (NJ012RM) was found to have a defective fuel rod. Since the Cycle 20 core design includes re-insertion of this assembly, the remaining, non-defective fuel rods were recaged into a new assembly and a steel pin was used to replace the defective rod. The steel pin was placed within the recaged assembly in a location that minimizes its impact on core reactivity. The recaged assembly was evaluated for its impact on the overall core design, found to be acceptable and certified by Areva for use in Cycle 20. The upper end fitting of the “failed” assembly was reused on the recaged assembly to maintain the core loading pattern presented in the RR.

Operating limits, setpoints, protective limits, and physics parameters are very similar to previous cycles. Independent methodology calculations confirmed that physics parameters were reasonable. All analyses methodologies used for Cycle 20 are approved by the NRC and are referenced in BAW-10179P-A, “Safety Criteria and Methodology for Acceptable Cycle Reload Analyses”, or as noted in the COLR or LOCA Summary Report.

Screening Basis:

The Cycle 20 Reload Report describes and addresses the design, accident analyses, and limiting operating conditions for the ANO-1 Cycle 20 core. All cycle-specific technical specification limits and setpoints for operation of Cycle 20 are placed in the COLR as allowed by USNRC Generic Letter 88-16. Technical Specification 5.6.5 requires the use of the latest NRC-approved Areva topical report BAW-10179P-A, “Safety Criteria and Methodology for Acceptable Cycle Reload Analyses.” All analysis methodologies used for Cycle 20 are approved by the NRC and referenced in the latest revision of BAW-10179P-A, or as noted in the COLR or LOCA Summary Report.

Operating License Impact? **NO**

As stated in the Cycle 20 Reload Report, the reload analyses and recommended operating limits and setpoints fall within the requirements for operating the ANO-1 core as described or referenced in the ANO-1 Operating License. The specific results of the reload analyses are beyond the scope of the NRC Confirmatory Orders. The Cycle 20 core design and results fully comply with the criteria discussed in the ANO-1 Technical Specifications. Note however that insertion of Mark-B-HTP fuel into the ANO-1 core required NRC approval of two Technical Specification changes, one to TS 2.1.1.2, the other to TS 4.2.1. The NRC approved both of these TS changes via their SER accompanying Amendment 226. All the input assumptions and methods are consistent with or conservative with respect to the current ANO-1 Technical Specifications (including Amendment 226). All cycle-specific limits for operation of the Cycle 20 core are located in the Cycle 20 Core Operating Limits Report (COLR). Furthermore, current Technical Specification safety limits, limiting safety settings, and limiting conditions of operation are bounding for the Cycle 20 core. Therefore, no further changes to the ANO-1 Operating License, Confirmatory Orders, or Technical Specifications are required to support the operation of the Cycle 20 core with regards to the reload analyses.

Impact on FSAR/LBD's Controlled by 50.59? **YES**

Each cycle, the RR is produced with the intent of replacing the contents of Chapter 3A of the SAR in its entirety. As such, a SAR change is required for Chapter 3A and the Master Table of Contents. Additionally, with the insertion of Mark-B-HTP fuel assemblies into the ANO-1 core, changes to Chapters 1, 3, and 14 are required. Most of the Chapter 14 changes are driven by a change to the LOCA analyses due to the introduction of the Mark-B-HTP fuel assemblies, as summarized in the Reload Report, as well as the LOCA Summary Report.

The COLR must be updated to reflect the Cycle 20 Reload Report limits and setpoints, as well as changes to the ICDNB surveillance requirements and refueling boron concentration.

Insertion of Mark-B-HTP fuel assemblies into the ANO-1 core requires changes to Technical Specification bases B 2.1.1, B 3.2.5 and B 3.4.1. The Technical Specification bases describe dose calculations associated with the Steam Generator Tube Rupture, Main Steam Line Break, and Loss of Load events, but the assumptions stated in the Technical Specification bases for these events remain unchanged. The results of the Cycle 20 reload analyses and the operating limits are also within the remaining requirements established by the Technical Specification bases. Thus, the information presented in the Technical Specification bases, with the exception of the three bases noted above, remains valid with regards to the Cycle 20 reload analyses.

All changes to the SAR, COLR and Technical Specification bases are described in the LDCR being presented with this Cycle 20 Reload Report 50.59 Review and will be addressed in the evaluation portion (Section VI) of this review. The change to the SAR Master Table of Contents is administrative in nature, does not change the scope of the SAR discussion and will not be discussed further.

The results of the Cycle 20 reload analyses and the operating limits and setpoints are beyond the level of detail present in the ANO-1 Technical Requirements Manual (TRM) and do not result in invalidating any information presented in the TRM.

The Cycle 20 reload analyses and operating limits and setpoints as stated in the RR and COLR, and the LOCA analyses inputs and results as summarized in the LOCA Summary Report, are within the requirements described in the ANO-1 NRC Safety Evaluation Reports (SERs) for operation of the ANO-1 core. The Cycle 20 reload and LOCA analyses do not invalidate any information presented in the NRC SERs.

Impact on LBDs Controlled Under Other Regulations? **NO**

The results of the Cycle 20 reload analyses and the operating limits and setpoints stated in the Reload Report, COLR and LOCA Summary Report are beyond the scope of the Fire Hazards Analyses (FHA). Cycle 20 calculations have demonstrated that there will be sufficient reactor coolant system (RCS) boration due to makeup for the RCS shrinkage during cooldown to have safe-shutdown capability. This fact is noted in the FHA with the cycle-specific Physics Manual listed as a reference. In addition, the remaining LBD's controlled under other regulations (QAPM, Eplan, etc.) are unaffected by the results of the Cycle 20 reload analyses. The Cycle 20 reload analyses and the changes to operating limits and setpoints associated with the reload analyses are beyond the level of detail present in those documents.

Does the Proposed Activity Involve a Test or Experiment Not Described in the SAR? **NO**

Implementation of the Cycle 20 core design and reload analyses does not constitute testing or experimentation. All characteristics of the Cycle 20 core are determined using NRC approved methods. Tests will be performed to verify design characteristics of the Cycle 20 core, but these tests are the same as last cycle, such that no changes are necessary to what is already described and approved within the SAR.

Does the Proposed Activity Potentially Impact Equipment, Procedures, or Facilities Utilized for Storing Spent Fuel at an ISFSI? **NO**

The Cycle 20 reload does not change the design bases of the spent fuel pool or associated support systems. The dry fuel storage selection procedures contain specific requirements that must be met in order for fuel assemblies to be stored in dry casks. Depending on operation of Cycle 20 and future cycles, the Cycle 20 fuel assemblies should be able to meet the requirements established in those procedures. In addition, no new requirements for storage of spent fuel at the ISFSI are necessary due to this reload design and introduction of the new Mark-B-HTP assemblies. Thus, the Cycle 20 reload does not impact equipment, procedures, or facilities utilized for storing spent fuel at an ISFSI. A 72.48 review has been completed and is included as part of this 50.59 review.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

Electronic search method used:	Keywords:
AUTONOMY/LRS 50.59-UNIT-1	ALL (reload*, core* NEAR1 design*, fuel* NEAR1 design*, burnup*, imbalance*, cycle* NEAR5 19, cycle* NEAR5 20, mtc*, moder* NEAR1 coef*, moder* NEAR1 temp*, fuel* NEAR3 *press*, temperat* NEAR5 reduct*, bypass* NEAR10 flow*, quad* NEAR1 power* NEAR1 tilt*, incore* NEAR1 detect*, short* NEAR1 emitt*, long* NEAR1 emitt*, radial* NEAR5 peak*, pin NEAR5 peak, peak* NEAR5 factor*, power NEAR1 peak, dropped* NEAR5 rod*, energy NEAR1 deposition, *edf*, stainless* NEAR1 steel* NEAR3 rod*, clad* NEAR1 strain*, fuel* NEAR5 swell*, LHR*, linear* NEAR1 heat* NEAR1 rate*, enrichment*, shutdown* NEAR1 margin*, rem, rcs* NEAR5 flow*, offset NEAR1 slope, incore* NEAR10 excore*, rod NEAR1 worth NEAR5 test, physic* NEAR1 man*, deplet*, control NEAR1 rod NEAR10 worth, group NEAR5 5, batch*, PCT, peak NEAR1 clad NEAR1 temp*, core NEAR10 hydrogen)
LBDs reviewed manually:	
ANO-1 SAR	All of Chapters 1, 3, 6, & 14
ANO-1 TS & Bases	ALL
ANO-1 COLR	ALL
ANO-1 TRM	ALL

5. Is the validity of this Review dependent on any other change?

☐ **Yes**

☒ **No**

If "YES," list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any new or different chemicals that are currently not authorized for use by the state regulatory agency?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117 for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following question if one of questions C.1 through C.10 above was answered "yes."

Is a change to the Security Plan required? ☐ **Yes** ☐ **No**

Attach to this 50.59 Review or reference below documentation for accepting a "yes" answer for any of Questions C.1 through C.10, above.

Name of Security Plan reviewer (print / Signature / Data

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

(NOTE: This section is not applicable to Grand Gulf or Waterford 3 and may be removed from 50.59 Reviews performed for Waterford 3 proposed activities.)

If any of the following questions is answered "YES," a 72.48 Review must be performed in accordance with NMM Procedure LI-112 and attached to this 50.59 Review.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the ISFSI including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☒	☐	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power that could potentially impact cask loading or storage activities?
9.	☐	☒	Involve a change to the Fuel Building ventilation that could potentially impact cask loading or storage activities?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☒	☐	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Involve fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components power that could potentially impact cask loading or storage activities?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.

☐ Yes
☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

For the accidents evaluated in the SAR, the only events that have an initiator that could be affected by the reload core design presented in the Reload Report for Cycle 20 are (a) the Stuck In/Stuck Out/Dropped Rod event, (b) Fuel Loading Errors, and (c) the Fuel Handling Accident. As explained below, none of these accidents are affected by the Cycle 20 reload design.

The Cycle 20 reload core design has been found to meet all design criteria based on NRC approved methodologies including the new Mark-B-HTP assemblies. BAW-10179P-A fuel assembly design criteria includes the requirements (1) that a path for control rod insertion is ensured even for an assembly with the maximum credible damage, including a Safe Shutdown Earthquake; (2) that the holddown springs be capable of maintaining fuel assembly contact with the lower support plate during normal operation; and (3) that guide tube buckling not be allowed during normal operation and any transient condition where control rod insertion is required by the Safety Analyses. Fuel rod bow to the point of contact with the guide tube where guide tube deformation could occur is also precluded by reload design. In addition, the control rod assembly will not be able to be disengaged from the fuel assembly guide tubes during operation. All assembly burnups will stay well within licensed limits. No new phenomena are expected that could impact assembly strength, corrosion performance, fretting behavior, etc. The new Mark-B-HTP designed assemblies provide additional margin against corrosion performance and fretting behavior by using materials and a different design that have been proven by performance at other utilities. Moreover, the Cycle 20 reload design did not initiate changes to the control system, motors, etc. associated with the CRDM system. All CRDMs will be changed out to Type C, but that change has been evaluated in a separate 50.59 review. Drop rod testing will also be performed to ensure Technical Specification requirements, and thus analysis assumptions, are met. Therefore, the frequency of occurrence of a Stuck In/Stuck Out/Dropped Rod event will not be increased by the employment of the Cycle 20 core design as presented in the Reload Report.

Fuel assembly identification will continue to be prominently displayed on the upper end fitting for core loading verification prior to startup. Existing operating procedures require verification of the final core loading. In addition, startup physics testing will confirm that the core is responding as designed and thus is properly loaded. No changes to the Areva fabrication process have occurred that would cause a pin misload. Therefore, the probability of gross fuel assembly misplacement in the core due to the Batch 22 assemblies and Cycle 20 core design is not increased.

Although the upper end fitting of the Mark-B-HTP assembly is changed from a helical holddown spring design to a six-leaf holddown spring design, fuel assembly handling is not changed with the new design. No changes have occurred that would negatively impact the strength or dimensions of the assembly, or the assembly interfaces with the fuel handling equipment, storage racks, and neighboring assemblies. No physical fuel handling equipment changes are required due to the fresh assemblies. Thus, the frequency of occurrence of a fuel handling accident remains unchanged.

No changes to plant equipment or operating procedures, other than setpoints, are required to implement the Cycle 20 core design. As discussed above there are no impacts to any of the accident initiators due to the fuel assembly and core design changes. Therefore, the changes will not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the SAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

Equipment important to safety which could be impacted by the Cycle 20 reload core design includes: (a) control rods and their drive mechanisms, (b) axial power shaping rods and their drive mechanisms, (c) RCS safety-related instrumentation (e.g., incore detectors, RTDs, etc.) and (d) the reactor vessel. As explained below, none of the systems or components mentioned are impacted by the Cycle 20 core design. Thus, the Cycle 20 core design does not increase the probability of a malfunction of a SSC important to safety.

Fuel assembly design criteria assure that the reload core design will not impact the proper function of the control rods, axial power shaping rods, or their respective drive mechanisms. Cycle 20 operational characteristics will be very similar to Cycle 19. Thus, operating pressures, temperatures, neutron fluxes, etc. will remain within the design parameters for RCS safety-related instrumentation as in Cycle 19. Likewise, the continued use of past operating characteristics and parameters that are bounded by current safety analyses maintains the plant response to abnormalities or accident conditions within the parameters used as design bases for engineered safety features. Cycle 20 also continues the use of a low-leakage core design that will minimize fluence to the reactor vessel. Therefore, the projected end-of-cycle-20 fluence is bounded by the fluence used to develop the reactor vessel P-T curves.

In addition, there are no changes in equipment, nor is the function or duty of the equipment important to safety affected by introducing the Batch 22 reload fuel. There are no additional loads imposed on systems, structures, or components important to safety. System redundancy and independence is not affected. Support system performance is not degraded. No changes to the failure modes of the equipment important to safety were assumed in the reload analyses. Therefore, the implementation of the Cycle 20 core design as documented in the Reload Report, COLR and LOCA Summary Report will not result in an increase of the probability of a malfunction of a SSC important to safety.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The transient response of the plant to the abnormalities and accident scenarios analyzed in SAR chapter 14 will not be altered by the implementation of the Cycle 20 reload core design. All associated accident initiators and any single-failure equipment malfunctions postulations remain valid with respect to their impact upon accident analyses consequences. In addition, the peak fuel rod internal pressure will be below the maximum allowable value permitted for both in-reactor and spent fuel conditions.

Table 3A-6 of the Cycle 20 Reload Report documents the results of the dose calculations based on the Cycle 20 core design and compares them to Cycle 19 results and the SAR. This table is reproduced in the supplemental information section (attached) along with the values specified in the NRC SERs for ANO-1 as being acceptable. The slight changes in the dose consequences of the accident analyses are related to the use of the Cycle 20-specific radionuclide sources calculated from the actual Cycle 20 core design and postulated irradiation history. Any dose increases are minimal and not a result of changes in dose release scenario assumptions dictated by the accident scenario and the associated plant response. The values in the table demonstrate that although some of the predicted doses for the Cycle 20 core design increase slightly, all doses remain well below the acceptable SER doses. The Cycle 20 doses are equal to or less than the Cycle 19 doses in 13 of the 20 cases. In two of the seven “unbounded” cases (Control Rod Ejection Accident 30-day LPZ whole body and LOCA 30-day LPZ thyroid), the Cycle 20 doses are less than the analysis-of-record doses reported in the SAR. For each of the five remaining, unbounded cases, the increases over the Cycle 19 doses are far less than 0.1 rem, as well as less than 10% of the difference between the current analysis-of-record doses reported in the SAR and the applicable regulatory limits (the NRC SER allowable doses shown in the table). In addition, the increase in consequences over the Cycle 19 doses is so small it could be reasonably concluded that the consequences have not actually changed. Thus, per the 10CFR50.59 Review Program Guidelines, DG-LI-101, Rev. 5, all Cycle 20 dose increases are considered minimal increases.

Acceptance criteria have been established to preclude the occurrence of criticality in the fuel storage racks during normal or accident conditions. Criticality analyses have been performed on the spent and temporary fuel storage racks to ensure that the established criticality criteria are met at ANO-1. Based on those analyses, the irradiated Batch 22 fuel may be safely stored either in a cross–hatch configuration (3–of–4) in the Spent Fuel Storage Rack region 1 poison racks or in a checkerboard configuration (2–of–4) in the region 2 burnup racks. Because the Batch 22 fuel is within the established criticality constraints, the consequences of a criticality event are not increased for Cycle 20.

The Cycle 20 core design does not affect the function of equipment designed to control the release of radioactive material and does not result in a new pathway to the outside environment. This change does not impact any fission product barrier or challenge any fuel safety limits. The calculated post-accident doses will not impact access to vital areas and mitigating actions will not be impeded during a postulated accident. No assumptions previously made in evaluating consequences need to be modified. Therefore, the implementation of the Cycle 20 core design, as documented in the Reload Report, COLR and LOCA Summary Report, will not result in more than a minimal increase in the consequences of an accident previously evaluated in the SAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The introduction of the Batch 22 fuel during Cycle 20 operations does not change any assumption concerning equipment availability or failure modes, and does not require new equipment or a change in operating procedures, other than setpoints. No greater reliance is placed on a specific system, structure, or component to perform its design function as a result of this change. Thus, the transient response of the plant to abnormalities and accident scenarios analyzed in SAR Chapter 14 will not be altered by the implementation of the Cycle 20 core design and as such, all associated accident initiators and any single-failure equipment malfunction postulations remain valid with respect to their impact upon the accident analyses. Specifically, no change will occur in the radiological release rate/duration, no new release mechanisms can be postulated, and no impact will occur to any radiation release barriers.

The minimal changes in the dose consequences of the accident analyses (see "Basis" for question 3 above) are related to the use of Cycle 20 specific radionuclide sources calculated from the Cycle 20 core design and postulated irradiation history. These changes are not a result of changes in dose release scenario assumptions dictated by the associated plant response to an accident. The results of the analyses determine that the behavior of the plant is consistent with that predicted by previous analyses, so that the predicted conditions under which the equipment important to safety is being required to operate do not appreciably change. Changes to the COLR do not require any changes to the assumptions concerning equipment availability or changes. Therefore, the changes considered here will not result in more than a minimal increase in the consequences of a malfunction of a SSC important to safety previously evaluated in the SAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The Reload Report, including its summary of the LOCA Summary Report, concludes that by the examination of Cycle 20 core thermal, thermal-hydraulic, and kinetics properties, this core reload will not adversely affect the ability to operate the ANO-1 plant safely during Cycle 20.

There is no new equipment associated with the introduction of the Batch 22 fuel. The changes described in the Reload Report will not alter the way in which the plant operates, that is, no new operating conditions or plant configurations are created that might lead to a new or different type of accident. No new failure modes are created through implementation of the Cycle 20 core design.

The transient evaluation of Cycle 20 is bounded by previously accepted analyses. The key safety analysis parameters for Cycle 20 are bounded by the assumptions in the SAR analyses and/or subsequent cycle analyses. No initiator to any of the accidents described in the SAR was impacted. In addition, the frequencies of accidents previously thought to be non-credible have not been increased by the Cycle 20 core design. COLR changes are consistent with safety analyses assumptions. Therefore, the possibility for an accident of a different type than any previously evaluated in the SAR is not being created.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

There is no new equipment associated with the introduction of the Batch 22 fuel. The changes described in the Reload Report will not alter the way in which the plant operates, that is, will not create new operating conditions or plant configurations that might lead to different result should a malfunction of a structure, system, or component important to safety occur. No new failure modes are created through implementation of the Cycle 20 reload design. Since the results of the reload analyses have determined that the behavior of the plant is consistent with that predicted by previous analyses, no new condition during transients is being predicted which might require a different accident mitigation path. Likewise, no new systems, beyond those already credited, are challenged for accident mitigation. COLR changes are consistent with safety analysis assumptions. Therefore, a possibility of a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the SAR is not created by implementing the Cycle 20 core design.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?

☐ Yes
☒ No

BASIS:

As mentioned in the general reload information (section II.B), the analysis-based LOCA linear heat rate limits for the Batch 21 fuel had to be reduced slightly for the Cycle 20 core design. The reduction of LOCA LHR limits was necessary to comply with the limitation on power peaking factors at the limits of normal operation as specified in the SER to BAW-10192P-A. Thus, the reduction of LOCA LHR limits is in accordance with approved NRC methodologies and maintains the level of conservatism necessary for protection against exceeding the design basis limit for a fission product barrier. With this reduction in LHR limits, the peak clad temperatures calculated for the LOCA (Small and Large Break) are within the acceptable values designated in 10CFR50.46 and no fuel failure was predicted for any SBLOCA consistent with past analyses' success criteria.

The Cycle 20 analyses were performed using NRC-approved methodologies and conservative plant specific inputs provided in the ANO-1 Cycle 20 Safety Analyses Groundrules document (CALC-A1-NE-2004-02). All thermal-hydraulic, neutronic, and safety analyses demonstrated that acceptance limits approved by the NRC are not exceeded. Likewise, all mechanical design criteria were found to be met for Cycle 20. Consequences of Chapter 14 accidents were bounded by the SER limits. COLR limits ensure that safety analyses assumptions are preserved. Therefore, the changes necessary to implement the Cycle 20 reload design do not result in exceeding or altering the design basis limits for fission product barriers as described in the SAR.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☒ No

BASIS:

In accordance with Technical Specification 4.2.1, the Cycle 20 core was designed and evaluated using NRC-approved analysis methodologies under an approved quality assurance program. These methods are described in the reload topical for B&W plants, BAW-10179P-A, Rev.6, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses". BAW-10241P-A, "BHTP DNB Correlation Applied with LYNXT," is incorporated into BAW-10179P-A as the acceptable methodology for DNB analysis of Mark-B-HTP fuel. Although this is a new methodology applicable to ANO-1 (i.e., is not yet identified in the SAR), its use at ANO-1 has already been reviewed and approved by the NRC. There is no departure from any of the NRC-approved methodologies used in the thermal-hydraulic, fuel mechanical, safety analyses, or neutronics calculations. However, an inconsistency in the LOCA methodology referenced in BAW-10179P-A is discussed below for completeness.

The approved methodology for LOCA analysis is described in BAW-10192A, commonly called the Evaluation Model (EM). The EM states that SBLOCAs should not go through DNB, and the maximum break size for the SBLOCA spectrum was set at 0.75-ft^2 based on the BWC CHF correlation for the Mark-BZ grid (and also the BWCMV correlation for mixing vane grid fuel assemblies). However, the BHTP CHF correlation has a different formulation than the BWC CHF correlation and was implemented in a more conservative fashion to support LBLOCA applications. The BHTP CHF correlation predicted early DNB for the Mark-B-HTP fuel for the maximum SBLOCA break size of 0.75-ft^2. A revised SBLOCA approach for the BHTP CHF correlation was then developed to remove the additional conservatism in the correlation implementation. This approach only addresses the prediction of initial DNB and does not influence the cladding temperature calculations later in the transient (during the core uncovering phase). These improvements in the BHTP CHF model were implemented along with a more realistic RCS flow of 106.5 percent of design flow (versus the 102 percent of design flow) based on the EOTSG with 5% SGTP. When applied, break sizes of 0.5 ft^2 and smaller did not go through DNB initially for the Mark-B-HTP fuel. The 0.75 ft^2 case did go through DNB initially for the Mark-B-HTP fuel, which means it should be placed in the transition LOCA spectrum using large break methodology. Therefore, the Mark-B-HTP SBLOCA spectrum is based on a smaller range of break sizes ($\leq$0.5 ft^2) than the Mark-B9 SBLOCA spectrum ($\leq$0.75 ft^2).

The EM states, "Breaks with cross-sectional areas greater than 0.75 ft^2 demonstrate early cladding DNB. Breaks from this size up to and including twice the cross-sectional area of the largest reactor coolant pipe are therefore considered to be in the large break spectrum." Even though the 0.75 ft^2 break from the EM perspective is stated to be in the small break spectrum, the fact that it went through DNB initially means it must be considered in the large break spectrum. This designation is consistent with the definition of LBLOCA for the purpose of using LB methodology as opposed to SBLOCA methods given in the EM. Therefore, even though the EM states that the 0.75 ft^2 break is in the small break spectrum, based on the given reasoning in the EM, it is placed in the large break spectrum for the BHTP fuel, and it is not considered a deviation from approved methods. These results are consistent with implementation of Mark-B-HTP fuel at Crystal River. These results will be identified and presented to the NRC via the annually submitted LOCA report required by 10CFR50.46.

Therefore, the implementation of the Cycle 20 core design as described in the Reload Report and COLR does not result in or require a departure from a method described in the SAR or other LBD.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

Supplemental Information

Dose Assessment:
The table below presents the information found in Table 3A-6 of the Cycle 20 Reload Report. The allowable doses based on ANO-1 NRC SERs are also presented below for comparison and convenience.

COMPARISON OF SAR, CYCLE 19, AND CYCLE 20 ACCIDENT DOSES (REM)

Accident	SAR	Cycle 19	Cycle 20	NRC SER
Fuel Handling (Outside RB)				
2 hr EAB Thyroid	10.4	10.4	10.38	75
2 hr EAB Whole-body	0.3	0.151	0.15	6
Fuel Handling Inside R.B.				
2 hr EAB thyroid	69.1	69.2	69.22	75
2 hr EAB whole body	0.3	0.202	0.202	6
Steam Line Break				
2 hr EAB thyroid	1.6	1.78	1.776	30
2 hr EAB whole-body	0.0	0.009	0.0086	2.5
Steam Generator Tube Failure				
2 hr EAB thyroid	4.64	7.29	7.276	30
2 hr EAB whole-body	0.125	0.31	0.306	2.5
Control Rod Ejection Accident				
2 hr EAB thyroid	6.266	7.21	7.208	75
2 hr EAB whole-body	0.012	0.007	0.007	6
30 day LPZ thyroid	5.025	5.79	5.785	75
30 day LPZ whole-body	0.009	0.005	0.0052	6
Loss of Cooling Accident				
2 hr EAB thyroid	7.01	3.81	3.807	300
2 hr EAB whole-body	0.0165	0.03	0.0302	25
30 day LPZ thyroid	2.66	1.91	1.912	300
30 day LPZ whole-body	0.0106	0.02	0.023	25
Maximum Hypothetical Accident				
2 hr EAB thyroid	148.68	154.07	154.072	300
2 hr EAB whole-body	4.66	5.30	5.296	25
30 day LPZ thyroid	52.38	71.00	71.002	300
30 day LPZ whole-body	1.54	1.89	1.89	25

ANO 50.59 Evaluation Number

2005-034

I. OVERVIEW / SIGNATURES

Facility: ANO-1

Document Reviewed: _ ER-ANO-2002-0640-000, Polar Crane Uprate_ **Change/Rev.:** _ERCN 5

System Designator(s)/Description:

Description of Proposed Activity:

ERCN 5 will allow the temporary use of the work platform to remove the two 25 ton chain hoists (tuggers) from the ceiling of the reactor building. The tuggers weigh approximately 2400 lbs each and while being removed are considered to be a heavy load in accordance with ANO's compliance with NUREG-0612, *Control of Heavy Loads*. As described in the ERCN 5, this will be accomplished by use of a Genie lift stationed and secured to the polar crane work platform. An evaluation of the potential targets was conducted to determine if any SSCs that provide a safety function during Mode 6 could be impacted.

The evaluation revealed that two Reactor Building Coolers, decay heat piping, and service water piping to the coolers are located in the area below the tuggers. All but the two Reactor Building Coolers were found to be protected by concrete floors of at least 8 inches thickness located above them. A RB cooling train consists of two coolers and their associated fans and there associated SW lines for a total of 4 cooling units. The concrete floors are deemed capable of protecting the piping from the 2400 pound dropped weight of the hoist. The two exposed Reactor Building Coolers are not required due to the limited number of fuel assemblies in the reactor vessel. The two outside coolers and the SW lines are protected by the concrete floors at elevation 401'-6" providing any heat removal necessary.

Check the applicable review(s): **(Only the sections indicated must be included in the Review.)**

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-034)**	**Sections I, II, and IV required**

Preparer: Steve A Bennett/ **ORIGINAL SIGNED BY STEVE BENNETT** / EOI/Licensing/12-02-05
Name (print) / Signature / Company / Department / Date

Reviewer: Doyle G Adams/ **ORIGINAL SIGNED BY DOYLE ADAMS** / EOI/Design Eng/12-02-05
Name (print) / Signature / Company / Department / Date

OSRC: J.R. Eichenberger / **ORIGINAL SIGNED BY J.R. EICHENBERGER** / 12-03-05
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See LI-101 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	SAR 9.6.1.7.1
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III OR perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the LBD change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113.			

[1] If "YES," see LI-101. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 evaluation.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☐ **Yes**
☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113, if applicable. If obtaining NRC approval, document the change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.

3. Basis

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

OL/TSs - The OL/TSs do not discuss control of heavy loads.

TS 3.9.5 states that in MODE 6, with the water level < 23 feet above the top of the fuel seated in the reactor vessel, two DHR loops must be Operable and one DHR loop must be in operation to provide removal of decay heat. This requirement will not be changed.

The only time that the TSs (TS 3.6.5) require Reactor Building Cooling systems to be Operable is during Modes 1 through 4. .

FSAR - SAR Section 9.6.1.7.1 discusses the cranes that are credited for performing heavy load lifts onsite and their compliance with NUREG-0612. Based on the manner that the tuggers will be removed, the temporary configuration is not considered as a crane or similar lifting device that is described in the SAR. Since the activity will result in only a one-time short duration heavy lift, it will not considered necessary to meet all of the general requirements contained in Section 5.1.1 of NUREG-0612. The primary requirement that needs to be adhered to is the requirement to have a safe load path for assuring that either fuel or credited safety functions are protected. If safe load paths cannot be assured then a load drop assessment is necessary to assure that safety functions are maintained. Based on the targets identified from a potential load drop, the only system that could be directly impacted is one of the redundant reactor building coolers.

There is no permanent change to our compliance to NUREG-0612 or needed revision to the FSAR. Therefore, no Licensing Basis Document Change Request is necessary.

Test or Experiments - The activity to remove the tuggers does not constitute a test or experiment in accordance with 50.59.

The risk considerations for possible failure of the work platform under ERCN-5 are covered under a separate 10CFRF50.69 assessment.

4. **References**

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

Electronic search method used:	Keywords:
LRS – ANO-1 50.59	Heavy load*, decay heat, building cool*

LBDs reviewed manually:

FSAR 9.6.1.7.1

TS 3.9.5 and 3.6.5

5. **Is the validity of this Review dependent on any other change?**

☐ **Yes**

☒ **No**

If "YES," list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any new or different chemicals that are currently not authorized for use by the state regulatory agency?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117 for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following question if one of questions C.1 through C.10 above was answered "yes."

Is a change to the Security Plan required? ☐ **Yes** ☐ **No**

Attach to this 50.59 Review or reference below documentation for accepting a "yes" answer for any of Questions C.1 through C.10, above.

Name of Security Plan reviewer (print / Signature / Data

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

(NOTE: This section is not applicable to Grand Gulf or Waterford 3 and may be removed from 50.59 Reviews performed for Waterford 3 proposed activities.)

If any of the following questions is answered "YES," a 72.48 Review must be performed in accordance with NMM Procedure LI-112 and attached to this 50.59 Review.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the ISFSI including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power that could potentially impact cask loading or storage activities?
9.	☐	☒	Involve a change to the Fuel Building ventilation that could potentially impact cask loading or storage activities?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Involve fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components power that could potentially impact cask loading or storage activities?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.	☐ Yes ☒ No

Does the proposed Change:

1.	Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?	☐ Yes ☒ No

BASIS:

THE NEAREST TARGET FROM A POTENTIAL DROP OF A TUGGER IS THE CONTAINMENT COOLERS. THE DH LINE IS SUFFICIENTLY PROTECTED TO NOT BE IMPACTED BY A LOAD DROP. THE LOSS OF DECAY HEAT COOLING OR LOSS OF CONTAINMENT COOLING WHILE IN MODE 6 ARE NOT SAR CHAPTER 6 OR 14 ANALYSED ACCIDENTS. THEREFORE, THE POTENTIAL FOR DROPPING OF A LOAD IN EXCESS OF 2000 LBS ON THE DECAY HEAT LINES OR CONTAINMENT COOLERS DOES NOT IMPACT ANY ANALYSED ACCIDENTS.

2.	Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?	☐ Yes ☒ No

BASIS:

THE TUGGERS WILL BE LOWERED BY A HOIST WHILE USING A GENIE LIFT SECURED TO THE POLAR CRANE WORK PLATFORM. THE LOAD WILL BE TAKEN DIRECTLY FROM THE LOCATION WHERE THE TUGGER IS CONNECTED TO THE DOME AND DOWN TO THE GRATING/FLOOR BELOW. THE TUGGERS WILL BE LOCATED NEAR BUT NOT ABOVE THE DECAY HEAT LINES WHICH RUN ON THE WEST SIDE OF THE D-RING. THEREFORE, IT HAS BEEN DETERMINED THAT AN EXPECTED LOAD DROP WILL NOT IMPACT THE INTEGRITY OF THE DH SYSTEM. HOWEVER, THE TUGGERS ARE JUST ABOVE THE EXPOSED REACTOR BUILDING COOLERS. WHILE REMOVING THE TUGGERS, THE TUGGERS COULD UNDERGO A DROP FROM ABOVE THE POLAR CRANE GIRDERS TO THE TOP OF THE EXPOSED COOLERS. WITH FUEL IN THE VESSEL, EVEN THOUGH THE HEAT LOAD IS VERY SMALL, THE SOPPS REQUIRE ONE COOLER INCLUDING SERVICE WATER LINE TO BE AVAILABLE. IN MODE 6 THERE IS NO NEED FOR MULTIPLE COOLING TRAINS TO BE REQUIRED. THE COOLERS AND SW LINES ARE ADEQUATELY SEPARATED THAT NO MORE THAN ONE COOLER CAN BE IMPACTED BY A SINGLE TUGGER LOAD DROP. AT LEAST ONE SET OF THE PROTECTED COOLERS WILL BE AVAILABLE DURING THIS PERIOD. A LOAD DROP ANALYSIS IS NOT NECESSARY TO ANALYSIS THE DROP OF A SINGLE SET OF TUGGERS UNDER THE CONDITIONS OF THE ERCN. THEREFORE, THE ADEQUACY OF THE DH REMOVAL SYSTEM AND THE REACTOR BUILDING COOLING SYSTEM ARE ASSURED DURING MODE 6 OPERATION.

3.	Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR?	☐ Yes ☒ No

BASIS:

THERE ARE NO CHAPTER 6 OR 14 ACCIDENT ANALYSES THAT ARE APPLICABLE DURING MODE 6 OPERATIONS. THE FUEL THAT HAS BEEN MOVED TO THE CORE IS WELL AWAY FROM THE LOCATION THAT THE TUGGER REMOVAL WILL OCCUR AND THERE WILL BE NO FUEL MOVMENT DURING THE EVOLUTION EVALUATED UNDER ERCN-5. THEREFORE, THE DOSE CONSEQUENCES OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE FSAR WILL NOT BE CHANGED BY THE LOWERING OF THE TUGGERS ALLOWED BY ERCN 5.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

THE DH SYSTEM WILL NOT BE IN THE LOCATION OF A POTENTIAL TUGGER LOAD DROP. SINCE NOT MORE THAN ONE OF EITHER OF THE BUILDING COOLERS COULD BE IMPACTED, AND THAT THERE IS NO CREDIBLE CONDITION THAT FUEL IN THE REACTOR VESSEL COULD BE IMPACTED, THERE WILL BE NO EVENT THAT WILL CREATE A CONDITION WHERE AND INCREASE IN DOSE OR DOSE CONSEQUENCE. A PROTECTED COOLER WILL BE AVAILABLE AT ALL TIMES DURING THE TUGGER REMOVEAL. THEREFORE, THERE IS NO MORE THAN A MINIMAL INCREASE OF CONSEQUENCES DUE TO A MALFUNCTION OF AN SSC.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

THE ACCIDENT EVALUATED IN SAR CHAPTER 6 AND 14, ARE THOSE ASSOCIATED WITH NORMAL MODE 1 OPERATION. EVENTS THAT CAN OCCUR DURING MODE 6 ARE COVERED BY OUR COMPLIANCE WITH GENERIC LETTER (GL) 88-17 AND THE ANO-1 SOPPS. GL 88-17, "LOSS OF DECAY HEAT REMOVAL," DISCUSSES EVENT DURING SHUTDOWN MODES OF PLANT OPERATION, PROVIDES RECOMMENDATIONS, AND REQUESTS INFORMATION FROM LICENSEES PERTAINING TO (1) PREVENTION OF ACCIDENT INITIATION, (2) MITIGATION OF ACCIDENTS BEFORE THEY POTENTIALLY PROGRESS TO CORE DAMAGE, AND (3) CONTROL OF RADIOACTIVE MATERIAL IF A CORE DAMAGE ACCIDENT SHOULD OCCUR. OUR COMPLIANCE WITH THIS GL AND THE ABILITY TO MITIGATE ANY EVENTS WILL NOT BE CHALLENGED.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

THE FAILURE MODE OF A DROP OF THE TUGGERS HAS BEEN DETERMINED TO ONLY IMPACT ONE OF THE EXPOSED REACTOR BUILDING COOLERS FOR EITHER ONE OF THE TUGGER REMOVALS. IN MODE 6 ONLY ONE COOLER IS REQUIRED TO BE AVAILABLE WHEN FUEL IS IN THE REACTOR VESSEL. THEREFORE, THE RESULT OF A DROP OF A TUGGER WILL NOT CHANGE THE RESULTS OF A MALFUNCTION PREVIOUSLY EVALUATED IN THE FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

THE DECAY HEAT LINES AND THE REACTOR BUILDING COOLERS ARE NOT PART OF ANY THREE FISSION PRODUCT BOUNDARIES WHICH ARE FUEL INTEGRITY, RCPB, OR THE CONTAINMENT. THERE IS NO CREDIBLE HEAVY LOAD FAILURES THAT COULD IMPACT ONE OF THE FISSION PRODUCT BOUNDARIES. THEREFORE, THE ACTIVITIES BEING PERFORMED IN ERCN-5 DO NOT IMPACT THE SAR DESCRIBED REQUIRMENTS OF THESE BOUNDARIES.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

THE REQUIREMENTS TO COMPLY WITH NUREG-0612 DO NOT CONSTITUTE A METHOD OF EVALUATION AS DEFINED BY 10CFR50.59. THEREFORE, THE PROPOSED CHANGE TO PERFORM A HEAVY LOAD LIFT AND THE POTENTIAL DROP OF A LOAD IN CONTAINMENT DOES NOT IMPACT ANY METHOD OF EVALUATION DESCRIBED IN THE FSAR.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2006-001

I. OVERVIEW / SIGNATURES

Facility: ANO-1

Document Reviewed: ______ER-ANO-2004-0944-000__________________ **Change/Rev.:** __0___

System Designator(s)/Description: D – 125VDC

Description of Proposed Activity:
ER-ANO-2004-0944-000 was developed to evaluate the impact of elevated specific gravity on exiting TS voltage and specific gravity limits and seismic integrity of the station batteries.

CR-ANO-1-2003-00553 documents elevated nominal specific gravity on both the Red and Green Train battery banks. These batteries are rated at a nominal specific gravity of 1.215. The Red Train's (D-07) average specific gravity is 1.225 and the Green Train's (D-06) average specific gravity is 1.235. During evaluation of this condition, it was identified that the basis for existing Tech Spec Table 3.8.6-1 voltage and specific gravity limits, are nominal specific gravity values of 1.215. Since CR-ANO-1-2003-0553 documented elevated specific gravities; CR-ANO-1-2004-01834 was initiated to document this condition – specifically elevated specific gravities may impact Tech Spec voltage and specific gravity limits. ER-ANO-2004-0944-000 was developed to evaluate the impact of this condition on existing TS voltage and specific gravity battery cell limits and the impact that this condition (elevated specific gravity) has on the seismic ability of the batteries.

Check the applicable review(s): **(Only the sections indicated must be included in the Review.)**

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: ___________________________)**	**Sections I, II, and IV required**

Preparer: Adrian Meyer / **ORIGINAL SIGNED BY ADRIAN MEYER** / EOI / Sys Eng / 3/13/06
Name (print) / Signature / Company / Department / Date

Reviewer: Alan Smith / **ORIGINAL SIGNED BY ALAN SMITH** /EOI / Sys Eng / 3/13/06
Name (print) / Signature / Company / Department / Date

OSRC: Thomas A. Marlow / **ORIGINAL SIGNED BY TOM A. MARLOW** / 3/16/06
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See LI-101 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☐	☒	
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III OR perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the LBD change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113.			

[1] If "YES," see LI-101. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 evaluation.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR?

☐ **Yes**
☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change <u>AND</u> initiate an LBD change in accordance with NMM LI-113, if applicable. If obtaining NRC approval, document the change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.

3. <u>Basis</u>

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

Tech Spec Table 3.8.6-1 identifies voltage and specific gravity limits for an operable battery. SR 3.8.6.1 and 3.8.6.2 require that the limits established in this table are verified on a weekly (pilot cell test) and quarterly (all cells) basis. Tech Spec Basis section Table 3.8.6-1 provides a basis for the limits established in this table. ER-ANO-2004-0944-000 concludes that existing TS Table 3.8.6-1 voltage & specific gravity limits continue to ensure the station batteries are capable of performing their safety function when operated within these voltage and specific gravity limits. This ER also concludes that the seismic capability of the batteries remain bounded by their electrical capability and therefore no change to the Tech Specs are required or recommended.

This condition has no impact on the SAR. An electronic and manual review of the SAR was performed as documented below.

ER-ANO-2004-0944-000 does not authorize any tests or experiments. This ER has been developed to evaluate the impact of elevated specific gravity on the ability of Tech Spec Table 3.8.6-1 voltage and specific gravity limits to protect the ability of the station batteries to perform their safety function. The conclusion of the Engineering Evaluation is that the Tech Spec limits will continue to protect the integrity of the batteries ensuring their ability to perform their safety related function of supplying their 2 hour duty cycle while maintaining their bank voltage greater than or equal to 105V.

4. <u>References</u>

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

Electronic search method used:

Keywords: Specific Gravity; Electrolyte

Autonomy - U1-50.59; U1-License Orders; U1-Original SER; U1-FSAR; E-Plan; ODCM; VSC-LDBs

LBDs reviewed manually:

TS 3.8.4, 3.8.6, B3.8.4; B3.8.6

FSAR 8.3.2 (DC Power Systems); 14.1.2.8 (Loss of Electric Power); 14.2.2.5 (Loss of Coolant Accident); Table 14-1 (Abnormalities Affecting Core and Coolant Boundary); Table 14-8 (Situations Analyzed for Safeguard Analysis)

5. Is the validity of this Review dependent on any other change? ☐ **Yes** ☒ **No**

If "YES," list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any new or different chemicals that are currently not authorized for use by the state regulatory agency?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117 for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following question if one of questions C.1 through C.10 above was answered "yes."

Is a change to the Security Plan required? ☐ **Yes** ☐ **No**

Attach to this 50.59 Review or reference below documentation for accepting a "yes" answer for any of Questions C.1 through C.10, above.

Name of Security Plan reviewer (print / Signature / Data

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

(NOTE: This section is not applicable to Grand Gulf or Waterford 3 and may be removed from 50.59 Reviews performed for Waterford 3 proposed activities.)

If any of the following questions is answered "YES," a 72.48 Review must be performed in accordance with NMM Procedure LI-112 and attached to this 50.59 Review.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the ISFSI including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power that could potentially impact cask loading or storage activities?
9.	☐	☒	Involve a change to the Fuel Building ventilation that could potentially impact cask loading or storage activities?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Involve fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components power that could potentially impact cask loading or storage activities?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

A. Check the applicable box below. If a box is checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function:

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists. Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof.
Reference: ____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions.

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

A review of the accidents evaluated in the Unit 1 FSAR Chapter 14 was performed. Chapter 14 provides a discussion of the "Abnormalities Affecting Core and Coolant Boundary" (Table 14-1) which includes a Loss of Electric Power event and "Situations Analyzed for Standby Safeguards Analysis" (Table 14-18) that includes a Loss of Coolant Accident. The Loss of Coolant Accident assumes offsite power is lost at time of event and a loss of one EDG as the postulated single active failure. These are the only two "events" addressed in the accident analysis evaluated in Chapter 14 that credit performance of the station batteries to mitigate the consequences of the events. Neither of these two events identifies failure of either station battery (D06 or D07) as the initiator of either event. The condition evaluated by this safety evaluation is "elevated specific gravity" of the station batteries which will not lead to either of these two events. Failure of a battery would affect the ability of the associated train to mitigate the two events addressed in the FSAR (Loss of Electric Power & DBA LOCA) but would not initiate either event. Since failure of the station batteries is not postulated to initiate accidents analyzed in the FSAR, the condition addressed by this safety evaluation will not increase the frequency of occurrence of an accident previously evaluated in the FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The batteries are sized with a significant amount of "additional margin" above safety margins that are recommended by IEEE 485 (Battery Sizing). Margins recommended by the IEEE standard include aging (25%), temperature (11%) and design (10%-15%). The batteries are LCR-21 type C&D Technology batteries. These batteries contain 10 positive plates (pp). Per battery sizing calc (92-E-0021-01, Rev.9) the bounding battery size is required to have 5.01pp associated with the Red Train battery (D07). The batteries contain > 25% margin of capacity above the margins recommended by the IEEE standard.

The IEEE working group that submitted existing TS specific gravity (SG) limits to the NRC credited capacity margin associated with the IEEE standard margins to allow specific gravity limits of 1.190 from a nominal 1.215 SG; sighting a 6% drop in battery capacity. Allowing the SG to drop from a nominal 1.235 to 1.190 would result in a drop in capacity of 13.5%. Capacity is a linear function of specific gravity (capacity drop of 3% for every 10 points of SG). Since the ANO-1 batteries contain an additional margin of > 25% capacity above that recommended by the IEEE standard and the basis for the allowable 1.190 SG credits margin provided by the IEEE standard; allowing the existing elevated specific gravity and existing TS specific gravity limits will not increase the likelihood of occurrence of a malfunction of a battery evaluated in the FSAR.

Voltage limits established in the TS of 2.13V and 2.07V protect the life of the battery and operability of the battery respectively. Allowing a battery to remain undercharged for an extended period of time leads to loss of life as a result of lead sulfate forming on the cell plates. Allowing a battery cell voltage to drop below 2.07V allows the cell to self-discharge thus minimizing its capacity leading to questionable ability to deliver its required amp-hour load thus requiring the battery to be declared inoperable.

A cell's recommended minimum allowable float voltage (2.13V) is based on its fully charged open circuit voltage (OCV) plus an additional "polarization potential" required to prevent self-discharge and maximize life. A cell with a nominal specific gravity of 1.215 has a nominal open circuit voltage of 2.063; a minimum allowable vendor voltage of 2.13V and vendor recommendation to initiate a equalize charge at a voltage of 2.12V. A cell with a nominal specific gravity of 1.235 has a nominal open circuit voltage of 2.083; a minimum allowable voltage of 2.15V and a recommended equalize voltage of 2.14V. Instructions contained in the battery TD allow for received batteries to be stored in an open circuit condition for up to 6 months with guidance to perform inspections on a monthly basis providing a "boost charge" should their voltage and/or specific gravity drop by 20mV or 25 points from the fully charged open circuit values. OP-1307.063 (Station Battery Surveillance Procedure) requires an equalize charge placed on a cell whose voltage is < 2.17V. This procedure also requires the bank voltage to remain between 127.6V (2.20vpc) and 130.5V (2.25vpc). The bank's voltage is verified every 7 days and each cell's voltage is verified every 92 days. Service discharge tests are conducted on the battery every 18 months and Performance discharge tests every 60 months. These tests verify the battery's ability to deliver its required amp-hour load and capacity or age respectively. Results of these tests demonstrate the batteries are being maintained in a fully charged state and are aging at a "normal" expected rate. Adequate controls are provided by OP-1307.063 to ensure minimum required voltages are maintained on the battery cells to prevent premature aging.

To preserve the same amount of margin as the current TS limit of 2.07V provides, a limit of 2.09V would be appropriate for a battery with a specific gravity of 1.235. This limit is designed to protect the operability of a battery with a cell self-discharging caused by internal cell shorting. Based on review of past discharge tests a cell's voltage initially collapses well below 2.0V (coup-de-fouet) and then slowly returns to a voltage just under 2.0V and again drops linearly as its amp-hours are depleted. A cell's voltage drops as its stored energy is depleted. The difference between existing TS limit (2.07V) and 2.09V is 0.02V. This difference in voltage results in a loss in capacity of approximately 7% (0.02V ÷ (2.09V–1.81V) X 100%). The <u>capacity depleted from a cell</u> prior to having to declare the battery inoperable remains bounded by the <u>additional margin</u> contained within the battery, above its <u>safety margin</u>. Therefore maintaining the current TS operability limit of 2.07V will continue to ensure the battery has sufficient capacity to perform its function.

NUREG/CR-5448 "Aging Evaluation of Class 1E Batteries: Seismic Testing" concludes that a battery's seismic capability remains bounded by its electrical capability as a result of its natural aging process that affects both its capacity and seismic ruggedness. If a battery is maintained in accordance with IEEE 450 then its seismic capability will be maintained. The maintenance procedure (OP-1307.063) is based on IEEE 450-1995. Calculation 88-E-0034-55 addresses the seismic qualification for the batteries and contains C&D report No. QR-283738-01. This report documents the aging mechanism of a battery cell is oxidation of the positive plate grid structure due to charging current. As the grid structure oxidizes its resistance increases reducing the battery's capacity and the grid's mechanical strength.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The accidents evaluated in the FSAR that credit the station batteries in mitigating the accident are a Loss of Electric Power event and a Loss of Coolant accident. Should the battery(s) fail during either of these postulated events, the consequences of the event would increase as a result of failure of the equipment that is dependent on the function of the battery(s) (EDG, SWGR BRK Operation, EFW Valves, Inverters, etc.).

The condition addressed by this safety evaluation (elevated specific gravity of station batteries) does not affect the battery's ability to perform its function. Per EPRI TR-100248, sulfuric acid (H2SO4) within the electrolyte and the active material of the plates (Pb (-) & PbO2 (+)) are “depleted” as a cell converts its stored chemical energy into active electrical energy. During this chemical reaction, the electrolyte is converted to water and the plates are converted to lead sulfate. Eventually the quantity of active material (SO4, Pb and PbO2) is depleted to a point where the cell’s capacity is depleted; reflected in a reduced terminal voltage (1.81 Vpc for a 58 cell bank or 105V bank voltage). Since the condition addressed by this safety evaluation is elevated specific gravity, thus elevated active SO4, this condition will not adversely affect the capacity of the battery; that is, the battery's safety function to provide its duty cycle required amp-hour of load.

The batteries are being maintained in accordance with vendor and industry guidance to maximize the battery's life - the rooms are maintained at approximately 77 °F, the float voltage is maintained within the acceptable float voltage range and discharge testing is limited to less than once every 12 months. These batteries are significantly oversized for their application. Per calc (92-E-0021-01, Rev.9) the batteries contain an “additional margin” factor of 25% above recommended safety margin sizing factors recommended by IEEE 485 (Battery Sizing).

Also during each of their service discharge tests (1R15, 1R16, 1R18 & 1R19) they have consistently demonstrated a high margin of capacity (end of test terminal voltage >113V) above their required minimum capacity (end of test terminal voltage of 105 V). Both batteries demonstrated a capacity of > 100% during their "Modified Performance Discharge Test" performed during 1R17 and neither battery showed signs of aging. These batteries were replaced during 1R14 (Spring 1998).

Discharge testing required per SR3.8.4.2 (18 months) and 3.8.4.3 (60 months) are used to detect when the battery's capacity begins to degrade, due to its natural aging process, and can be used to trend the battery's rate of aging thus allowing one to predict the batteries end of life. Also, current SR 3.8.4.3 requires increased discharge testing when the battery begins to show signs of degradation, based on its demonstrated capacity, during the discharge testing.

NUREG/CR-5448 “Aging Evaluation of Class 1E Batteries: Seismic Testing” concludes that a battery’s seismic capability remains bounded by its electrical capability as a result of its natural aging process that affects both its capacity and seismic ruggedness and that if a battery is maintained in accordance with IEEE 450 and Reg Guide 1.129 then its seismic capability will be maintained. The batteries surveillance procedure (OP-1307.063) is consistent with IEEE 450-1995 and in fact the TS Bases references this IEEE standard.

Therefore, since the batteries are maintained in a very "healthy environment," have demonstrated no signs of degradation, and testing requirements are in place and controlled through TS surveillances this condition will not increase the consequences of an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The batteries are designed to provide 125VDC power to their associated safety related loads for a 2-hour duration, in the event of a loss of AC power. Review of FSAR Chapter 14 events identifies two events that credit operation of the batteries in mitigating these events. They are <u>Loss of Electric Power</u> and <u>Loss of Coolant Accident</u>. The Loss of Coolant Accident assumes offsite power is lost at time of event and the loss of one EDG as the postulated single active failure. These are the only two "events" addressed in the accident analysis evaluated in Chapter 14 that credit performance of the station batteries to mitigate the consequences of the events.

The Loss of Electric Power credits the batteries for supporting the Turbine Drive EFW Train in mitigating this event. The Loss of Coolant accident credits the battery for starting the associated train's EDG. The Loss of Coolant accident postulates failure of one of the EDGs concurrent with a loss of offsite power. Postulating a failure of the battery that is required to support the remaining EDG would increase the consequences of this accident, however failure of two class 1E devices would have to be postulated which is beyond design basis. Elevated specific gravity will not adversely affect the performance of the batteries to deliver their designed amp-hour loads or affect the batteries' seismic capability.

The condition (CR-ANO-1-2003-0553) addressed by this safety evaluation has existed since initial installation of the batteries in spring 1998. Results of discharge testing demonstrate that the batteries have consistently demonstrated a high level of performance indicating that this condition has not adversely affected the performance of the batteries. This condition does not result in more than a minimal increase in the consequences of a failure of a SSC important to safety evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

Per review of Calc 87-E-0039-01, Rev. 1, (Hydrogen Generation of Batteries), hydrogen generation for a lead acid battery decreases with increasing specific gravity. Thus this condition (elevated specific gravity) will reduce the amount of hydrogen generation that these batteries produce. Hydrogen is the only toxic/flammable gas that is generated by these batteries and thus this condition does not create the possibility for an accident of a different type than any previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

Postulated failure of a battery is bounded by its inability to deliver its stored energy in the form of amp-hours at a terminal voltage of greater than or equal to 105 V to its safety related loads. Elevated specific gravity will not cause the batteries terminal voltages to be above existing "float voltage" conditions. The condition addressed by this safety evaluation will not affect the results of a postulated failure of a station battery and its impact on any events evaluated in the FSAR. Postulated failure of a station battery is limited to failure of the associated train's EDG, EFW DC control valves, switchgear breaker operation, inverters, etc. The results of the loss of one safety Train is currently addressed in the FSAR. This condition does not create the possibility of a malfunction of a SSC important to safety with a different result than any previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

This condition has no impact on any fission product barrier – fuel cladding, RCS or Containment integrity. The condition addressed by this safety evaluation is solely associated with the station batteries which support mitigation of events described in the FSAR, but do not affect the design basis limit for a fission product barrier as described in the FSAR.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

The batteries are sized in accordance with IEEE 485 to be capable of supplying their emergency duty cycle load for a 2-hour duration with a postulated battery at its end of life (80% capacity rating) and minimum allowed operating temperature (60 °F). The condition described in this safety evaluation does not affect the sizing methodology or design margins (aging & temperature) that are used in sizing the batteries. The condition addressed by this safety evaluation will not adversely affect the ability of the battery to deliver its required duty cycle load. A battery's capacity is a function of the quantity of sulfuric acid (electrolyte – H2SO4), lead dioxide (+ plate - PbO2), lead (- plate - Pb). The capacity of the battery is limited to the most limiting quantity of these elements. Since elevated specific gravity is an indication of increased sulfuric acid and has no impact on the other elements that affect the capacity of the battery, this condition will not adversely affect the ability of the batteries to perform their safety functions. This condition does not affect any method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

ANO 50.59 Evaluation Number

2006-002

I. OVERVIEW / SIGNATURES

Facility: ANO - Common

Document Reviewed: OP-1000.152 **Change/Rev.:** 004-01-0

System Designator(s)/Description: Unit 1 & 2 Fire Detection / Suppression Systems

Description of Proposed Activity:

Item #1: Revision of procedure OP-1000.152 to include fire zones 2102-Y, 2099-W, 2108-S, 2076-HH and 2068-DD on Attachment 2 Table 9D-1 "Fire Detection Instruments". Fire Protection Engineering recommended in response to CR-ANO-2004-00982-CA-9 that all fire zones where manual operator actions are required as identified in procedures OP-1203.049 and OP-2203.049 that the detection systems be added to the OP-1000.152 table of fire zones with applicable fire detection instruments. The review determined that fire zones 2102-Y, 2099-W, 2108-S, 2076-HH and 2068-DD were the only missing fire zones. There currently exist no regulatory requirement for inclusion of these zones in the fire system specifications, rather they are being added as a prudent measure to ensure that systems are available or other compensatory measures are in place if inoperable.

Item #2: Revise Unit 1 FSAR section 9D.3.3.B to read "If any Reactor Building Cable Spreading Area fire detection is inoperable per Attachment 1 section 9D.1.3.B, establish a fire watch patrol inside containment in the affected area once every 8 hours; or, monitor reactor building temperature at computer points T6278 and T6279 and document temperature hourly with a temporary log. Initiate fire response actions upon identification an unexpected increase in containment temperature. Additional indications may be, but are not limited to, equipment performance (motor amps, vibrations, etc.) and abnormal indication from installed instrumentation and controls." Revise Unit 2 FSAR section 9D.3.1.C to read "If any Containment Building Cable Spreading Area fire detection is inoperable per Attachment 2 section 9D.1.1.C, establish a fire watch patrol inside containment in the affected area once every 8 hours; or, monitor reactor building temperature at computer points T5605-5 and T5606-6 and document temperature hourly with a temporary log. Initiate fire response actions upon identification of an unexpected increase in containment temperature. Additional indications may be, but are not limited to, equipment performance (motor amps, vibrations, etc.) and abnormal indication from installed instrumentation and controls." These revised compensatory measures are added to the corresponding sections of OP-1000.152. Unit 1 & Unit 2 containment fire suppression systems are non-automatic systems due to the containment isolation valves being normally closed. Operator intervention is required upon identification of a fire in the containment cable spreading areas to align the systems for fire water flow into the building. As such, the containment suppression systems cannot be considered operable if no indication is available to indicate a fire. Upon declaring the suppression system inoperable a continuous fire watch is required. For ALARA reasons the current specification requirements are seen as unacceptable and inconsistent with industry practices for PWR plants. Exception to these compensatory measures is needed to allow an alternate method of verification of a fire inside containment until inoperable detection can be restored. ER-ANO-2006-0107-000 reviewed the HVAC system and available containment temperature instruments. The controls for operation of the containment fire suppression systems are unchanged, only the mechanism for identifying a fire and the response time to a fire. The design function of the detection is to notify control room operators of a fire in the containment cable spreading areas. The revised compensatory measures have been identified as industry wide standards found acceptable by the NRC for other PWR containment buildings.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☒	**SCREENING (Applies to item #1)**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: ____________________________)** **(Applies to item #2)**	**Sections I, II, and IV required**

Preparer: Ronald D. Hendrix / **ORIGINAL SIGNED BY RONALD D. HENDRIX** / ENS / EP&C / 3-30-06
Name (print) / Signature / Company / Department / Date

Reviewer: Jackie L. Johnson / **ORIGINAL SIGNED BY JACK L. JOHNSON** / ENS / EP&C / 4-3-06
Name (print) / Signature / Company / Department / Date

OSRC: J. N. Miller, Jr. / **ORIGINAL SIGNED BY J. N. MILLER, JR.** / 4-17-06
Chairman's Name (print) / Signature / Date
(Required only for Programmatic Exclusion Screenings and 50.59 Evaluations.)

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES," obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See LI-101 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	U1 FSAR 9D.3.3.B / U2 FSAR 9D.3.1.C
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES," perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the LBD change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☒	☐	(Reference: U1 FSAR 9D.3.3.B / U2 FSAR 9D.3.1.C)
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES," evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113.			

[1] If "YES," see LI-101. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 evaluation is performed. Attach the 50.54 evaluation.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. Does the proposed activity involve a test or experiment not described in the FSAR? ☐ **Yes** ☒ **No**

If "YES," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113, if applicable. If obtaining NRC approval, document the change in Section II.A.5. However, the change cannot be implemented until approved by the NRC. Complete Section II.

3. Basis

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR. If the proposed activity involves a potential test or experiment not previously described in the FSAR also include an explanation. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

Item #1: Revision of Detection and Suppression System Applicability Tables:

Operating License / Technical Specifications

The identification of Appendix R detection and suppression system operability and surveillance requirements is below the scope of operating license documents.

FSAR

The FSAR contains a listing of regulatory required detection and suppression systems. The systems in the FSAR are those identified in each unit's original Technical Specifications and later moved verbatim to the FSAR as permitted by generic letter 86-10. Procedure 1000.152 is now the implementing procedure for detection and suppression system operability and surveillance requirements as delineated in each FSAR. The addition of other systems to this procedure does not require revision of the FSAR as they are not required by regulation to be included into the surveillance program. The identified additions are being included as a measure to ensure detection out of service time is minimized in area where fires result in operator manual actions. There are no other LBD's impacted by this change.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

Electronic search method used:	Keywords:
Autonomy: 50.59-Common	fire specification*
LRS -50.59-Common	compensatory and fire

LBDs reviewed manually:

Unit 1 FSAR Appendix 9D;

Unit 2 FSAR Appendix 9D

5. Is the validity of this Review dependent on any other change? ☐ **Yes** ☒ **No**

If "YES," list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115 and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Involve a land disturbance equal to or in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve any land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, ditch, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any new or different chemicals that are currently not authorized for use by the state regulatory agency?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?[1]
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117 for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following question if one of questions C.1 through C.10 above was answered "yes."

Is a change to the Security Plan required? ☐ **Yes** ☒ **No**

Attach to this 50.59 Review or reference below documentation for accepting a "yes" answer for any of Questions C.1 through C.10, above.

Name of Security Plan reviewer (print / Signature / Data

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING
(NOTE: This section is not applicable to Grand Gulf or Waterford 3 and may be removed from 50.59 Reviews performed for Waterford 3 proposed activities.)

If any of the following questions is answered "YES," a 72.48 Review must be performed in accordance with NMM Procedure LI-112 and attached to this 50.59 Review.

Will the proposed activity being evaluated:

	YES	NO	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the ISFSI including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power that could potentially impact cask loading or storage activities?
9.	☐	☒	Involve a change to the Fuel Building ventilation that could potentially impact cask loading or storage activities?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Involve fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components power that could potentially impact cask loading or storage activities?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

A. Check the applicable box below. If a box is checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function:

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists. Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof.
Reference: ____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions.

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   Of the accidents evaluated in the FSAR, none of the accident initiators are related to a containment building fire or the ability to detect a fire. Therefore, changes to the compensatory measures for inoperable fire detection have no impact on the frequency of occurrence of an accident.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   General Design Criteria 3 "Fire Protection" states in part that fire detection and fighting systems of appropriate capacity and capability are provided and designed to minimize the adverse effects of fires on structures, systems, and components (SSC) important to safety. Those structures, systems, and components important to safety as applied to a fire are those required to achieve and maintain safe shutdown (SSD) in accordance with 10CFR50 Appendix R. The physical arrangement of the reactor buildings and the applicable SSC's provide adequate separation for required safe shutdown components. This separation assures that a fire of significant magnitude would be restricted to either the north or south fire zones in the containment buildings. The partial coverage fire detection and suppression systems in the cable spreading areas are not intended to limit the spread of a fire from north to south fire zones. Rather, these systems are intended to provide local protection of the cables concentrated in the penetration areas. Neither, the type of fire detection employed, nor changes to the frequency of inspection can initiate a fire. As such, no more than a minimum likelihood of malfunction due to a fire of a structure, system, or component important to safety previously evaluated in the FSAR could result from a delay in the response time to a fire when the proposed compensatory measures are employed.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   No accidents were identified that were previously evaluated in the FSAR for which fire detection and suppression is considered an accident mitigator. As such, the consequences of no accident previously evaluated in the FSAR will be increased.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The SSC's required for SSD are located in either the north and south fire zones for a fire in the opposite fire zone. Affected SSC's in the fire affected fire zone are all considered to be fire damaged resulting in worst case failure or spurious operation. Given that a fire cannot transport from one fire zone to another regardless of the response time associated with identification of a fire, then a delay in response to a fire condition when the proposed compensatory measures are in place will not damage any additional components not already considered in the Appendix R analysis. As such, the method of detecting a containment fire will not result in an increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   Fire damage is considered to result in worst case failure or spurious operation of SSC. Given this worst case damage fire is not considered to be an accident initiator. Therefore, response time to a fire will not cause a new type of accident from those previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   SSD components have been evaluated for worst case failure of spurious operation. Changing the response time to a containment fire does not cause a different result. Containment fires have been evaluated to determine worst case fire induced failures. These failures bound any possible damage that could cause a SSD component to malfunction. A train of required SSD components will be available in the fire zone on the opposite side of containment. No additional SSD will be damaged as a result of increased fire duration. An increased fire duration does not change the result of circuit failure analysis for those in the area subjected to fire damage.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

   BASIS:

   The existing Appendix R analysis insures that for a containment fire in either unit that specific performance goals are achieved. These goals including reactivity control, maintaining pressurizer level within operating level, decay heat removal, process instrumentation functional, etc. which maintain the integrity of the reactor coolant system and the fission product barriers. A fire in containment will not have any affect on the integrity of the containment boundary since the condition analyzed is already within the design analysis.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

   BASIS:

   The change of compensatory measures and potential response time change when the proposed compensatory measures are in place do not affect any method for evaluating design bases or in the safety analyses. Therefore no departure from a method of evaluation is created.

If any of the above questions is checked "YES," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure LI-113.

Enclosure 2

To

0CAN050607

ANO-1 and ANO-2 Commitment Change Summary Report

ANO Commitment Change Summary Report

Number	Commitment Description	Reason For Change
79	Develop Separate Audits For EOP/AOP (Biennial Audit)	The commitment being deleted relates to NRC identified weaknesses in planning/performing QA audits of emergency operating procedures in 1990. The changes were made to the site QA audit procedures for EOPs/AOPs (QAP-30) and the commitments were maintained until the procedure's deletion. The commitment was then transferred to the master audit plan (MAP) for the operations audit area. The MAP for operations has evaluation criteria for assessing EOPs. With the development of these newer tools and the fact that the activities that implement the subject commitments have been incorporated in QA processes for so long, tracking these activities by commitment is no longer necessary"
336	Ensure Future IST Program Updates Use GL 89-004 Part D [Commitment Was Made In The Second 10 Year Inservice Testing Program Submittal]	This commitment is no longer relevant to current program documents and is, therefore, deleted. (Reference: CR-ANO-C-2004-01129)
604	Implement Controls For Authorizing Unescorted Site Access By June 30, 1991	These commitments were implemented over 2 years ago and are now part of the AA/FFD normal and audited processes. Therefore, it is no longer necessary to track this item as a commitment.
683	Procedure 2102.010, Plant Shutdown and Cooldown, Requires RCS Structural Integrity Inspection Upon Plant Shutdown	Step 7.17 of 2102.010 directs the performance of the reactor building walkdown in hot standby. The note states, "at a minimum, RCP cavities (4.3.1) and pressurizer area should be inspected for leakage. 2102.001 Attachment D, RCS Structural Integrity Inspection, may be used as guideline for any additional areas to be checked". The complete "RCS Structural Integrity Inspection" is performed while in cold shutdown per step 9.17 of 2102.010. Therefore, it is no longer necessary to track this item as a commitment.
1119	Procedure 1000.120 Was Initiated to Control Breached Fire Barriers Until They Are Properly Sealed	The above commitment is captured in OP-1000.120. The above requirement is contained within federal regulation and is not required to be tracked as a commitment. OP-1000.152 also requires a firewatch to be posted for breached penetration seals.

ANO Commitment Change Summary Report

Number	Commitment Description	Reason For Change
2394	Revise Procedures if an Unfavorable Radiological Safety Evaluation Review Results	The origin of the commitment was NRC Violation, 313/9030 02, involving the inadequate review and documentation of the safety analysis for conducting resin transfer cask dewatering performed under the 10 CFR 50.59 procedure (OP-1062.004) in effect at that time. Full compliance with the required commitment action was achieved by September 30, 1991. This effort has continued to the present date, and every occurrence of the activities linked to the listed commitments has received specific review under the Radiological Safety Evaluations (RSE) process. RSEs conducted during the 15 years since the initiating event have found that these activities have no adverse impact on the site radiological release limits, plant protective boundaries or safety limits. In summary, the 1990 event which initiated the violation was an isolated occurrence. Full implementation of the commitment has been achieved and the programmatic aspects of the 10 CFR 50.59 process and RP work planning are adequate such that the commitments are no longer required to demonstrate compliance with the plant license or regulations.
2395	Revise Procedures if an Unfavorable Radiological Safety Evaluation Review Results	The origin of the commitment was NRC Violation, 313/9030 02, involving the inadequate review and documentation of the safety analysis for conducting resin transfer cask dewatering performed under the 10 CFR 50.59 procedure (OP-1062.004) in effect at that time. Full compliance with the required commitment action was achieved by September 30, 1991. This effort has continued to the present date, and every occurrence of the activities linked to the listed commitments has received specific review under the Radiological Safety Evaluations (RSE) process. RSEs conducted during the 15 years since the initiating event have found that these activities have no adverse impact on the site radiological release limits, plant protective boundaries or safety limits. In summary, the 1990 event which initiated the violation was an isolated occurrence. Full implementation of the commitment has been achieved and the programmatic aspects of the 10 CFR 50.59 process and RP work planning are adequate such that the commitments are no longer required to demonstrate compliance with the plant license or regulations.

ANO Commitment Change Summary Report

Number	Commitment Description	Reason For Change
2403	Revise Procedures if an Unfavorable Radiological Safety Evaluation Review Results	The origin of the commitment was NRC Violation, 313/9030 02, involving the inadequate review and documentation of the safety analysis for conducting resin transfer cask dewatering performed under the 10 CFR 50.59 procedure (OP-1062.004) in effect at that time. Full compliance with the required commitment action was achieved by September 30, 1991. This effort has continued to the present date, and every occurrence of the activities linked to the listed commitments has received specific review under the Radiological Safety Evaluations (RSE) process. RSEs conducted during the 15 years since the initiating event have found that these activities have no adverse impact on the site radiological release limits, plant protective boundaries or safety limits. In summary, the 1990 event which initiated the violation was an isolated occurrence. Full implementation of the commitment has been achieved and the programmatic aspects of the 10 CFR 50.59 process and RP work planning are adequate such that the commitments are no longer required to demonstrate compliance with the plant license or regulations.
2404	Restrict Processing of Radioactive Materials to Areas Inside Controlled Access or Invoke Additional Radioactive Controls	The origin of the commitment was NRC Violation, 313/9030 02, involving the inadequate review and documentation of the safety analysis for conducting resin transfer cask dewatering performed under the 10 CFR 50.59 procedure (OP-1062.004) in effect at that time. Full compliance with the required commitment action was achieved by September 30, 1991. Full implementation of the commitments has been achieved and the programmatic aspects of the 10 CFR 50.59 process and RP work planning are adequate such that the commitments are no longer required to demonstrate compliance with the plant license or regulations.
4811	Make Procedure Changes Establishing Appropriate Respiratory Protection During Trash Compaction	This commitment was written in response to NRC inspection violation 313/8706-01. The violation was based upon the failure to adequately monitor worker exposure to airborne radioactivity as required by 10 CFR 20.201(b) while worker's performed radioactive trash compaction. The compaction of radioactive trash is no longer performed at ANO. Radioactive trash is sent to an off-site vendor for reduction prior to disposal. The elimination of radioactive trash compaction makes this commitment no longer necessary

ANO Commitment Change Summary Report

Number	Commitment Description	Reason For Change
5118	Ensure Fire Watch Personnel With Appropriate Respiratory Equipment Stationed In Potentially Affected Area	This commitment is captured in OP-1000.120. The above requirement is contained within federal regulation and is not required to be tracked as a commitment. At the time of this commitment, a continuous fire watch was required for a breached fire barrier. Regulations have changed to a roving on one side if detection is operable thus there is probably no need to ever have to don respiratory equipment to inspect a breached fire barrier
5119	Roving Fire Watch With Respiratory Equipment Would Relieve Personnel in Affected Areas	Commitment is captured in OP-1000.120. The above requirement is contained within federal regulation and is not required to be tracked as a commitment. At the time of this commitment, a continuous fire watch was required for a breached fire barrier. Regulations have changed to a roving on one side if detection is operable thus there is probably no need to ever have to don respiratory equipment to inspect a breached fire barrier
5592	Safety System Walkdown Procedure is Being Developed	The items evaluated as per 1015.024 (Safety System Walkdown) are adequately evaluated by EN-DC-178 (System Walkdowns) and 1015.001 (Conduct of Operations) under watchstanding duties. This commitment has subsequently been captured as part of an on-going program or other administrative control that is subject to a revision review process and is, therefore, deleted.
5831	Request Relief For Reactor Coolant Pump Casing Weld Flaw Indications	This commitment was a result of ANO-1 seeking relief from ASME Section XI, Subsection IWB 2420 and IWA 2430 in the second 10 year inservice inspection (ISI) for "C" and "D" Reactor Coolant Pumps (RCPs) due to a flaw detected in the "A" RCP casing. This relief was approved with the condition that the licensee conduct an augmented inservice inspection program. A subsequent relief request was generated for relief from the augmented ISI program originally requested in lieu of a fracture mechanics evaluation per ASME Code Case N-481 (1CAN099302). This relief request was also approved by the NRC. ANO has performed all the required inspections as noted in NRC correspondence 1CAN099302. Included in this correspondence are further letters 1CAN069201 and 1CAN069207. This involved applying code case

Number	Commitment Description	Reason For Change
		N 481 which are alternate examination requirements for cast austenitic pump casings. The requirements are 1) performing a VT-2 inspection of the exterior of all pumps during the hydrostatic pressure test required by Table IWB 2500 1 category B P, 2) perform a VT-1 visual examination of the external surfaces of the weld of one pump casing, 3) perform a VT-3 visual examination of the interior surfaces when ever a pump is disassembled for maintenance and 4) perform an evaluation to demonstrate the safety and serviceability of the pump casing. This evaluation was performed by structural integrity and found that the Unit 1 RCPs were safe to operate for the forty year life of the plant and was accepted by the NRC. Code Case N 481 has been included in the Unit 1 ISI Program (CEP ISI 002) for future inspections upon RCP disassembly. In conclusion, this commitment is incorporated into the ANO 1 ISI program and does not require further tracking in the ANO commitment tracking system.
6242	Indicate Negative Consequences That Make Testing at Code Required Frequency Impractical	This commitment is no longer relevant to current program documents and is, therefore, deleted. (Reference: CR-ANO-C-2004-01129)
6364	Perform Alternative Test to Detect Leakage Of A Few GPM Once Per Refueling Outage During Hot Shutdown	This commitment is no longer relevant to current program documents and is, therefore, deleted. (Reference: CR-ANO-C-2004-01129)
6365	Full Stroke Exercise And Stroke Time CV-1000 Once Per Every 3 Months	This commitment is no longer relevant to current program documents and is, therefore, deleted. (Reference: CR-ANO-C-2004-01129)
6366	Test ERV Opening And Closing Setpoints Once Per Refueling Outage	This commitment is no longer relevant to current program documents and is, therefore, deleted. (Reference: CR-ANO-C-2004-01129)
6367	Full Stroke Exercise ERV Once Per Refueling Outage Using RCS Pressure To Confirm Opening And Closing	This commitment is no longer relevant to current program documents and is, therefore, deleted. (Reference: CR-ANO-C-2004-01129)

ANO Commitment Change Summary Report

Number	Commitment Description	Reason For Change
6685	Measure Primary Flow Each Fuel Cycle To Ensure Actual Flow Meets Or Exceeds Minimum Flow Used In Design Calculations	Commitment is deleted as it has been incorporated into reload documentation for each specific cycle.
7080	Change Test Procedure To Be More Explicit With Regard To Test Function	Implementation of standardized technical specifications deleted testing of main steam and main feed isolation valves during power operations due to potential for plant shutdown as the commitment was applied against online testing this aspect of the commitment is no longer applicable. Testing methodology also changed and only uses control room handswitches for the components, therefore the previous testing utilizing components of the steam line break isolation cabinet no longer exists. This commitment no longer applies and may be deleted. Currently approved testing methodology for offline testing is maintained in 1102.001 Supplement 3
7233	Operations Procedures Will Be Revised To Include Watchstanding In Reduced RCS Inventory Considerations	This commitment has been implemented in operating procedures which are subject to 10 CFR 50.59 review. In addition, extensive training is conducted and procedural controls are in place to address this. As this commitment is greater than 2 years old, is maintained by documents subject to 10 CFR 50.59 review, it is no longer necessary to maintain this additional administrative requirement.
7814	Note All Safety Systems Taken Out Of Service In The Station Log And The Status Board	This commitment has been implemented in operating procedures which are subject to 10 CFR 50.59 review. In addition, this is required by the LCO tracking record. As this commitment is greater than 2 years old, and is maintained by documents subject to 10 CFR 50.59 review, it is no longer necessary to maintain this additional administrative requirement.
7882	Develop Program Of Periodic Safety System Walkdowns	The items evaluated as per 1015.024 (Safety System Walkdown) are adequately evaluated by EN-DC-178 (System Walkdowns) and 1015.001 (Conduct of Operations) under watchstanding duties. This commitment has subsequently been captured as part of an on-going program or other administrative control that is subject to a revision review process and is, therefore, deleted.

ANO Commitment Change Summary Report

Number	Commitment Description	Reason For Change
8063	QC Superintendent To Determine Specific Monitoring Of Contract Work Activities; Contract Administration Procedure Will Include Notification Of QC When Activities Which Require Monitoring Commence	This commitment required no NRC submittal. It is more than two years old and there have been no known recurrences of the condition, therefore, in accordance with guidance given in LI-110, Attachment 9.4, Part V, it may be eliminated. Current process require the contract manager to notify site audits of all contracts for on-site services that are to be performed to contractor's QA program. This step allows for the insertion of hold and witness points as necessary.
8179	Revise Procedure Regarding The Duties And Authority Of The Shift Supervisor	The duties and authorities of the shift manager are based on industry wide and Entergy fleet best practices. These requirements are being moved to corporate Conduct of Operations (EN-OP-115) and Operations Expectations and Standards (COPD-001). As this commitment is greater than 2 years old, and has no technical basis, it is no longer necessary to maintain this additional administrative requirement.
8181	Revise Procedures Regarding Shift And Relief Turnover	The duties and authorities of the shift manager are based on industry wide and Entergy fleet best practices. These requirements are being moved to corporate Conduct of Operations (EN-OP-115) and Operations Expectations and Standards (COPD-001). As this commitment is greater than 2 years old, and has no technical basis, it is no longer necessary to maintain this additional administrative requirement.
8183	Implement Procedures To Limit Control Room Access During An Accident	The duties and authorities of the shift manager are based on industry wide and Entergy fleet best practices. These requirements are being moved to corporate Conduct of Operations (EN-OP-115) and Operations Expectations and Standards (COPD-001). As this commitment is greater than 2 years old, and has no technical basis, it is no longer necessary to maintain this additional administrative requirement.
8931	Prepare Draft Revision Of QAP-13 Following 1985 Audit Of Emergency Preparedness To Enhance Ap&L/State/Local Interface	The commitments being deleted relate to NRC identified weaknesses in planning/performing QA audits of emergency planning during the early/mid 1980s. The enhancements were added as commitments to the site QA audit procedures for emergency planning (QAP-13) and were maintained there until the procedure's deletion. Since that time, new corporate procedures and

Number	Commitment Description	Reason For Change
		processes have been implemented that describe the requirements for planning and performing audits of emergency planning. The MAP for emergency planning has replaced the guidance previously contained in the site QA procedure and provides significantly more detailed evaluation criteria for areas to be assessed during the e plan audits including mandatory audit scope elements in the areas described by these commitments. With the development of these newer tools and the fact that the activities that implement the subject commitments have been incorporated in QA processes for so long, tracking these activities as commitments are no longer necessary.
8944	ISI Coordinator To Notify QC Whenever An ISI Contractor Is To Commence NDE Work	This commitment required no NRC submittal. It is more than two years old and there have been no known recurrences of the condition, therefore, in accordance with guidance given in LI-110, Attachment 9.4, part V, it may be eliminated. Current process require the contract manager to notify site audits of all contracts for on-site services that are to be performed to contractor's QA program, which would include ISI NDE activities. This step allows for the insertion of hold and witness points as necessary.
9326	Implement Procedure To Establish Administrative Controls On Fire Barriers	This commitment is captured in OP-1000.152. The above requirement is contained within federal regulation and is not required to be tracked as a commitment.
9610	Update Guidance For Fire Watch Responsibility	This commitment is captured in OP-1000.120. The above requirement is contained within federal regulation and is not required to be tracked as a commitment.
9931	Follow Actions Provided In Submitting Safeguards Information	This is not a commitment as defined by LI-110. However, ANO has classified it as a commitment and Entergy has been following the guidance, in part, outlined in the generic letter. In 2004 Entergy received approval from the NRC to submit safeguards information electronically. This is a standard process and does not need to be carried as a commitment. To ensure the generic letter remains linked as the source document for the guidance, a reference to the generic letter will be included in the reference section of LI-106.

ANO Commitment Change Summary Report

Number	Commitment Description	Reason For Change
9958	Check The Status Of All Safety Related Fire Doors On Daily Log	This commitment is captured in OP-1000.120. The above requirement is contained within federal regulation and is not required to be tracked as a commitment.
10083	Fire Extinguisher Procedures Revised To Ensure Extinguishers Fully Charged	At one time a fire watch was required to have a fire extinguisher for transient combustibles but per OP-1000.047 that requirement is no longer valid. Additionally, repetitive task ensure that all extinguishers are charged. This was actually a life safety concern.
10153	Use Colored Fire Retardant For Easy Identification	This commitment is captured in OP-1000.047. The above requirement is contained within federal regulation and is not required to be tracked as a commitment.
10154	Transient Lumber Used In Safety Related Areas Will Be Treated With Colored Fire Retardant	This commitment is captured in OP-1000.047. The above requirement is contained within federal regulation and is not required to be tracked as a commitment.
10342	Develop And Implement QA Procedure To Evaluate Effectiveness Of Emergency Action Training For Various Functional Areas	The commitments being deleted relate to NRC identified weaknesses in planning/performing QA audits of emergency planning during the early/mid 1980s. The enhancements were added as commitments to the site QA audit procedures for emergency planning (QAP-13) and were maintained there until the procedure's deletion. Since that time, new corporate procedures and processes have been implemented that describe the requirements for planning and performing audits of emergency planning. The MAP for emergency planning has replaced the guidance previously contained in the site QA procedure and provides significantly more detailed evaluation criteria for areas to be assessed during the e plan audits including mandatory audit scope elements in the areas described by these commitments. With the development of these newer tools and the fact that the activities that implement the subject commitments have been incorporated in QA processes for so long, tracking these activities by commitment is no longer necessary.

Number	Commitment Description	Reason For Change
10709	2107.001 Contains Weekly Check Of Nitrogen Isolation Valve To Each Electrical Penetration And Verification That It Is Pressurized	This commitment has been implemented in operating procedures which are subject to 10 CFR 50.59 review. In addition, all electrical penetrations have been replaced since this commitment was made and operations configuration control management will ensure the valves stay aligned. As this commitment is greater than 2 years old, and is maintained by documents subject to 10 CFR 50.59 review, it is no longer necessary to maintain this additional administrative requirement.
10867	QA To Ensure Scope Of Annual Audit Will Include Verification Necessary To Assure Adequate Emergency Plan Implementation	The commitments being deleted relate to NRC identified weaknesses in planning/performing QA audits of emergency planning during the early/mid 1980s. The enhancements were added as commitments to the site QA audit procedures for emergency planning (QAP-13) and were maintained there until the procedure's deletion. Since that time, new corporate procedures and processes have been implemented that describe the requirements for planning and performing audits of emergency planning. The MAP for emergency planning has replaced the guidance previously contained in the site QA procedure and provides significantly more detailed evaluation criteria for areas to be assessed during the e plan audits including mandatory audit scope elements in the areas described by these commitments. With the development of these newer tools and the fact that the activities that implement the subject commitments have been incorporated in QA processes for so long, tracking these activities by commitment is no longer necessary.
10927	Control Of Maintenance Procedure Addresses Establishment Of Firewatch While Replacing Pipe Through Fire Barrier Penetrations	This commitment is captured in OP-1000.152. The above requirement is contained within federal regulation and is not required to be tracked as a commitment. Appendix R requires compensatory measures to be taken when fire barriers are breached. This is a former fire protection technical specification that is now contained in the FSAR Appendix 9D and OP-1000.152.
11332	Vendor Personnel Will Work Under The Cognizance Of Assigned ANO Personnel	This commitment required no NRC submittal. It is more than two years old and there have been no known recurrences of the condition, therefore, in accordance with guidance given in LI-110, Attachment 9.4, part V, it may be eliminated. Current ENS process, as defined in ENS-MP-106, details contract

Number	Commitment Description	Reason For Change
		manager responsibilities for successful completion of all contractor work activities. These responsibilities and steps include pre-job briefings and oversight of all contractor activities.
11787	Establish Controls And Limits On The Amount Of Allowed Combustibles In Safety Related Areas	The above commitment is captured in OP-1000.047. The above requirement is contained within federal regulation and is not required to be tracked as a commitment. The inspection is conducted IAW OP-1003.005.
13334	Procedures Were Revised To Clarify The Maximum Background Count Rate While Performing A Direct Frisk Survey	P-13334 is referenced in two documents 1063.034 "Radiation Protection Task Qualification Training" and 1012.018 "Administration of Radiological Surveys". The source document referenced in procedure 1063.034 is referenced incorrectly. The need to perform training is an enhancement to the source document identifying an additional action by the licensee; therefore, the placement of the source document as a reference is in error. The original commitment is tracked by procedure 1012.018. The original commitment has been in place for 17 years, has been adopted as the corporate standard and is, therefore, no longer necessary to track as a commitment.
14028	Revise The U-1 IST Plan And HES-07 To Reflect 1CNA099401 (See Commitment Text For Procedure 1102.001 Implementation)	This commitment is no longer relevant to current program documents and is, therefore, deleted. (Reference: CR-ANO-C-2004-01129).
14715	Perform An Annual Audit Of Training Documentation For Off-Site Emergency Response Personnel	The commitments being deleted relate to NRC identified weaknesses in planning/performing QA audits of emergency planning during the early/mid 1980s. The enhancements were added as commitments to the site QA audit procedures for emergency planning (QAP-13) and were maintained there until the procedure's deletion. Since that time, new corporate procedures and processes have been implemented that describe the requirements for planning and performing audits of emergency planning. The MAP for emergency planning has replaced the guidance previously contained in the site QA procedure and provides significantly more detailed evaluation criteria for areas to be assessed during the e plan audits including mandatory audit scope elements in the areas described by these commitments. With the development

ANO Commitment Change Summary Report

Number	Commitment Description	Reason For Change
		of these newer tools and the fact that the activities that implement the subject commitments have been incorporated in QA processes for so long, tracking these activities by commitment is no longer necessary.
15451	Revise Ops Procedure to Include Requirements of Maintaining Respiratory Qualification Current and of Having Required SCBA Eyewear Readily Available	The procedural controls for this requirement have been moved to COPD-001, operations standards and expectations. This requirement is controlled by training procedures, is trained on, is checked on a periodic basis by the review of crew training form, and is part of the ops quarterly procedure review acknowledgement. As this commitment is greater than 2 years old, and is controlled by multiple processes, it is no longer necessary to maintain this additional administrative requirement.
16075	Revise Procedures 5120.423 And 5120.426 To Clearly Define Limitations On The Options For Flow Rate Verification Of 2vef-14a/B And 2vef-15	The actions that resulted from this commitment have been captured in station procedures op 5120.423 and 5120.426. Since the commitment is greater than 2 years old and these actions are embedded in the station procedures, it is no longer necessary to track this item as a regulatory commitment. Additionally, future changes to this item will be controlled in accordance with 10 CFR 50.59. Therefore, the additional administrative control provided by tracking this item as a regulatory commitment is unnecessary.
16635	Determine Whether There Are Any Generic Implications Associated With Engineering-Related Performance Problems	The 2003 NRC Problem Identification & Resolution Inspection identified a generic issue with condition report (CR) initiation threshold at ANO. A detailed action plan to address that issue was completed under CR-ANO-C-2003-01080 has corrected this problem at ANO. EN-LI-102 guidance re-enforces accountability for initiating CRs and establishes an appropriate threshold for when they should be initiated. Therefore, this item need no longer be tracked as a commitment.
17433	Revise Procedures To Assure That Utilization Of Escorts Does Not Circumvent The Unescorted Access Process As Described In Response To B.3.B	These commitments were implemented over 2 years ago and are now part of the AA/FFD normal and audited processes. Therefore, it is no longer necessary to track this item as a regulatory commitment.

ANO Commitment Change Summary Report

Number	Commitment Description	Reason For Change
17435	Implement Permanent Guidance To Limit Employees With Temporary Unescorted Access As Described In Response To B.3.D	These commitments were implemented over 2 years ago and are now part of the AA/FFD normal and audited processes. Therefore, it is no longer necessary to track this item as a regulatory commitment.
17626	Complete Implementation Of Compensatory Measure B.1.1	These commitments were implemented over 2 years ago and are now part of the AA/FFD normal and audited processes. Therefore, it is no longer necessary to track this item as a regulatory commitment.
17135	Visually Inspect Plugs Each Refueling Outage To Ensure No Leakage Path Exists On Any Westinghouse Alloy-600 Ribbed Cold Leg Plugs	ANO-1 original steam generators (OTSGs) were replaced during refueling outage 1R19. This commitment is no longer relevant and is, therefore, deleted.