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Stephenie L. Pyle
Manager, Licensing
Arkansas Nuclear One

2CAN091103

September 16, 2011

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

SUBJECT: 10 CFR 50.59 Summary Report
Arkansas Nuclear One, Unit 2
Docket No. 50-368
License No. NPF-6

Dear Sir or Madam:

In accordance with 10 CFR 50.59(d)(2), attached is the Arkansas Nuclear One, Unit 2 (ANO-2) 10 CFR 50.59 summary report for the time period beginning March 23, 2010 and ending September 16, 2011. This report contains a brief description of changes to procedures, the facility as described in the ANO-2 Safety Analysis Report (SAR), the ANO-2 Technical Requirements Manual, and the ANO-2 Technical Specification Bases, for which a safety evaluation was conducted. 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-2 specific and those evaluations that may be common between ANO-2 and ANO, Unit 1, if any, is included in the attachment to this submittal.

If you have any questions or require additional information, please contact me.

Sincerely,

ORIGINAL SIGNED BY STEPHENIE L PYLE

SLP/dbb

Attachment: ANO-2 10 CFR 50.59 Summary Report

cc: Mr. Elmo E. Collins
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U. S. Nuclear Regulatory Commission
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NRC Senior Resident Inspector
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Attachment to

2CAN091103

ANO-2 10 CFR 50.59 Summary Report

ANO-2 10 CFR 50.59 Summary Report

50.59 #	Initiating Document	Summary
2010-001	EC-16240	Addition of Time Delay Relays to the opening circuitry of Containment Sump Recirculation Isolation Valves 2CV-5649-1 and 2CV-5649-2
2010-002	EC-20413	Addition of Zinc Injection Equipment to the Chemical and Volume Control System
2010-003	NRC Letter dated July 1, 2010	Addendum 1 to Topical Report WCAP-16500-P, Supplement 1, Revision 1, "Application of CE Setpoint Methodology for CE 16 x 16 Next Generation Fuel (NGF)"

Acronyms (where not specifically listed in the associated 5059)

ANO-2	Arkansas Nuclear One, Unit 2
CE	Combustion Engineering
COLR	Core Operating Limits Report
CS	Containment Spray
CVCS	Chemical and Volume Control System
DNBR	Departure from Nucleate Boiling Ratio
EC	Engineering Change
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
HPSI	High Pressure Safety Injection
LOCA	Loss of Coolant Accident
NEI	Nuclear Energy Institute
NGF	Next Generation Fuel
NPSH	Net Positive Suction Head
OSRC	Onsite Safety Review Committee
PAD	Process Applicability Determination
PSA	Probabilistic Safety Analysis
PWR	Pressurized Water Reactor
RAS	Recirculation Actuation Signal
RCPB	Reactor Coolant Pressure Boundary
RWT	Refueling Water Tank
SER	(NRC) Safety Evaluation Report
SSC	Structure, System, or Component
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report (or SAR)

ANO 50.59 Evaluation Number

2010-001

I. OVERVIEW / SIGNATURES[1]

Facility: ANO Unit 2 **Evaluation #** FFN-10-001 **/ Rev. #:** 0

Proposed Change / Document: Addition of Time Delay Relays to the opening circuitry of Containment Sump Recirculation Isolation Valves 2CV-5649-1 and 2CV-5649-2 / EC-16240

Description of Change:

EC-16240 adds time delay relays (62/5649-1 and 62/5650-2) and associated wiring to the valve open circuits of containment sump recirculation outlet isolation valves 2CV-5649-1 and 2CV-5650-2. The time delay relay will delay opening of the containment sump outlet valves on receipt of a Recirculation Actuation Signal (RAS) by 35 - 40 seconds. This delay will provide additional margin and provide further assurance that air ingestion due to vortex formation at the Refueling Water Tank (RWT) suction point will not occur thereby ensuring safety injection and containment spray pump performance will not degrade during the suction swap to the containment sump.

The ANO Unit 2 SAR section 6.2.3.2.2.2 describes the sequence of events when a RAS occurs as: "The RAS automatically opens the suction line valves from the containment sump, while closing the minimum flow line valves to the RWT. The RWT outlet valves begin closing at the same time, but have a stroke time which is much longer than the sump isolation valves and will not restrict the flow of water to the pumps before the sump outlet valves are opened."

The above description is adversely affected by the EC-16240 change since the opening of the containment sump isolation valves will not begin at the same time as the start of closing the RWT outlet valves.

The RWT outlet valve allowable closing times range from 66 - 90 seconds with 90 seconds being the maximum limit. The allowable opening stroke times for the containment sump isolation valves range from 17.4 - 25 seconds with 25 seconds being the maximum limit. ANO Unit 2 SAR Table 15.1.0-7 specifies the response time requirements for opening the containment sump valves as 145.0 seconds from onset of the Refueling Water Tank Low level. EC-16240 and supporting analysis demonstrates that sufficient overlap exists with a time delay of up to 40 seconds for opening of the containment sump isolation valves to ensure adequate pump performance exists during the transfer and that adequate submergence sufficient to prevent vortexing at the RWT suction exists.

This 50.59 evaluation is based on removal of the Temporary Modification implemented under EC-3340 which short stroked RWT outlet valves 2CV-5630-1 and 2CV-5631-2. This is identified as a prerequisite for the Return to Service of EC-16240.

Is the validity of this Evaluation dependent on any other change? ☐ **Yes** ☒ **No**

If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation 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.

Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval? ☐ **Yes** ☒ **No**

[1] Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

Preparer: James S. Rowe / See AS for signature / EOI / I&C Design Engr / See AS for date
Name (print) / Signature / Company / Department / Date

Reviewer: R. Eric Allen / See AS for signature / EOI / I&C Design Engr / See AS for date
Name (print) / Signature / Company / Department / Date

OSRC: Dale E. James / ORIGINAL SIGNED BY DALE E. JAMES / 05-08-10
Chairman's Name (print) / Signature / Date

OSRC-10-017
OSRC Meeting #

II. 50.59 EVALUATION

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 UFSAR? ☐ Yes ☒ No

   BASIS:

   The affected equipment for this change is the valve opening circuitry for the containment sump recirculation outlet valves 2CV-5649-1 and 2CV-5650-2. A time delay relay is added to the circuitry to delay opening of these valves on receipt of a RAS actuation (i.e. – low RWT level) by as much as 40 seconds. These valves serve as an accident mitigator for the loss of coolant accident by providing a recirculation source from the containment sump for the containment spray (CS) and high pressure safety injection (HPSI) pumps following depletion of the RWT inventory during the injection phase of the accident. These valves and their associated circuitry function only as an accident mitigator and therefore, have no impact on the frequency of occurrence of an accident previously evaluated in the UFSAR.

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 UFSAR? ☐ Yes ☒ No

   BASIS:

   The SSCs directly impacted by this change are the containment sump recirculation outlet valves 2CV-5649-1 and 2CV-5650-2. Equipment indirectly impacted by this change are the HPSI pumps (2P-89A, B, C) and containment spray pumps (2P-35A, B). The new time delay relays to be added to the opening circuitry of 2CV-5649-1 and 2CV-5650-2 meet the original design and construction practices. These new components and associated wiring are qualified as safety related components and are seismically qualified and mounted, and maintain divisional separation requirements.

   The mitigating function of the directly impacted devices described above are to automatically open following receipt of a RAS initiation thereby providing a non-restricted suction source to the HPSI and CS pumps. The opening sequence will still satisfy the SAR requirement for the containment sump isolation valves to open within 145.0 seconds of receipt of the RWT low level. Additionally, the time delay added has been selected such that sufficient overlap exists in the transfer between the two suction sources thereby maintaining sufficient NPSH to the CS and HPSI pumps during the duration of the transfer. This valve opening timing ensures the indirectly impacted equipment identified above will continue to meet their performance requirements.

   The containment minimum and maximum flood levels are not impacted by this change. Therefore, the evaluated net positive suction head (NPSH) available for the minimum flood level will not adversely impact the CS and HPSI pump performance. Additionally, since the maximum flood level is not impacted, no additional equipment located in containment will be submerged and the environmental qualification (EQ) of equipment required to function in the post accident environment is not affected.

   The new time delay relays do add a slight increase to the failure probability of the containment sump recirculation isolation valves to open. Should the time delay relay contacts fail to close following the energization of the relay coil; the associated containment sump recirculation outlet valve will fail to open. The probabilistic safety analysis (PSA) failure rate of a relay failing to energize is approximately 2.5E-05 failures per demand. The PSA failure rate for the containment sump recirculation outlet valves to open on demand is 9.995E-04 failures per demand. There is no impact to the single failure criteria since if one valve fails to open on demand, the redundant train is still available to satisfy the accident mitigating functions. Therefore, it is concluded that the addition of time delay relays does not significantly increase the probability of the valves to open on demand and is much less than the factor of 2 specified as the criteria in EN-LI-101 Rev 6. Thus, 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 (SSC) important to safety previously evaluated in the UFSAR.

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

☐ Yes
☒ No

BASIS:

The accident of concern with respect to this change is the Loss of Coolant Accident (LOCA) described in the UFSAR chapter 15.1.13. The impacted components are the containment sump recirculation isolation valves 2CV-5649-1 and 2CV-5650-2 and their respective trains of HPSI and CS which they support. These components and systems serve as accident mitigators and act to protect the fuel cladding and prevent fuel melting, limiting chemical reactions and to protect the health and safety of the public.

Upon receipt of a RAS, the containment sump recirculation isolation valves 2CV-5649-1 and 2CV-5650-2 have a delayed opening by as much as 40 seconds by their respective time delay relays 62/5649-1 and 62/5650-2. The 40 second time delay is selected to improve the margin to vortexing in the RWT during the transfer thereby providing sufficient net positive suction head (NPSH) while preventing the ingestion of air due to vortexing to the HPSI and CS pumps. SAR table 15.1.0-7 indicates the assumed response time for the containment sump recirculation valve opening from the time of the refueling water tank low level is 145.0 seconds. The stroke time of the RWT outlet valves (2CV-5630-1 and 2CV-5631-2) to close is a minimum of 66 seconds and a maximum of 90 seconds. The stroke time of the containment recirculation isolation valves to open is a minimum of 17.4 seconds and a maximum of 25 seconds. With the proposed 40 second time delay, the maximum time from receipt of the rwt low level (i.e. – RAS initiation) to the containment sump recirculation outlet valve being fully opened is 65 seconds. This is well within the 145 seconds assumed in the accident analysis. Based on the information above, it is concluded there is no adverse impact to the performance of the HPSI and CS systems and supporting components.

There are no impacts to the source term assumed in the accident analysis from the proposed change. Since there is no change to the source term and the performance of the mitigating systems is not adversely impacted by the addition of the proposed time delay, it is concluded there is no impact to offsite dose considered in USFSAR chapter 15.1.13 as a result of the change.

Based on the above discussion, it is concluded that the mitigating systems performance continues to perform within the bounds of the existing analysis and therefore, there is no increase in the consequences of an accident previously evaluated in the UFSAR as a result of the proposed change.

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 UFSAR?

☐ Yes
☒ No

BASIS:

The proposed addition of a time delay to the opening circuitry of containment sump recirculation isolation valves 2CV-5649-1 and 2CV-5650-2 does not create any new release pathways. Additionally, should a containment sump recirculation isolation valve fail to open (i.e. – single failure), the consequence of the failure to open would not be any better or worse than had the valve fail to open prior to the proposed change. UFSAR section 15.1.13 evaluated post accident leakage from the recirculation leakage, other gross leakage, and the consequences of an ECCS pump seal failure. Given the above discussions regarding the maintaining adequate performance of the mitigating systems, there are no new failure mechanisms or more severe leakage malfunctions introduced as a result of the proposed change. Therefore, it is concluded that there is no increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

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

   BASIS:

   The proposed addition of a time delay to the opening of the containment sump recirculation isolation valves on a receipt of a RAS initiation only has potential impacts to the subject valves and the HPSI and CS systems they support. Based on a review of the ANO Unit 2 UFSAR and the type of accidents evaluated, both of these systems are accident mitigators and are not accident initiators. Based on the scope of the change proposed, there is no new credible accident that would be possible as a result of implementation of the proposed change. Therefore, there is no possibility for an accident of a different type than previously evaluated in the UFSAR.

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 UFSAR? ☐ Yes ☒ No

   BASIS:

   The proposed addition of a time delay to the opening of the containment sump recirculation isolation valves on a receipt of a RAS initiation only has potential impacts to the subject valves and the HPSI and CS systems they support. Based on a review of the ANO Unit 2 UFSAR, the following sections address failures of these SSCs:

   1) Section 6.2.2.1.1 and Table 6.2-23 address the CS system and Containment Sump Isolation Valves
   2) Section 6.3.2.11.2, Section 6.3.2.11.4, and Table 6.3-3 address the HPSI system and Containment Sump Isolation Valves

   The above referenced sections of the UFSAR identify that the CS and HPSI systems are designed such that the systems ability to fulfill the safety related functions is not degraded by the failure of a single active component. The CS system single failure analysis considered the failure of the containment sump isolation valve to open on a RAS. The UFSAR requirement is that both valves on <u>either</u> suction header are required to open. The proposed time delay is added to only one of the two suction valves in each header (the other inboard isolation valve is normally open). The HPSI system single failure analysis considered the failure modes of the sump isolation valve as either inadvertently opening during use of the RWT during the injection mode or inadvertently closed during the use of the sump.

   The addition of the time delay in the open circuit of the containment sump recirculation isolation valves has no impact on the failure modes already considered in the UFSAR. The failure modes possible for the valves are failure to open when required, inadvertently open when not required, or fail closed during recirculation mode. All of these failure modes have already been evaluated in the UFSAR failure modes and effects analysis and no new failure modes are introduced as a result of this change. Therefore, the proposed change does not create a possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the UFSAR.

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

BASIS:

The 3 fission product barriers are 1) fuel cladding, 2) RCS pressure boundary (RCPB), and 3) Containment boundary. The proposed change is associated with adding a time delay to the opening circuits of the containment sump recirculation isolation valves 2CV-5649-1 and 2CV-5650-2. These valves receive a signal to open following a RAS which only occurs in a post LOCA environment. In the post LOCA environment, the reactor coolant pressure boundary must have been violated since it is the assumed accident therefore, consideration of maintaining the RCPB becomes moot for this proposed change and will not be addressed in answering this question. The two barriers of concern for this change are the fuel cladding and the containment boundary.

The UFSAR Chapter 15 accident analysis assumes that during a LOCA, the fraction of the cores radioactive inventory are released into containment are 100% of the noble gases, 25% of the iodines and 1% of all other fission products. Since at least some breach of the fuel cladding is considered in the post LOCA condition, the parameters of concern in answering this question is to ensure the HPSI system continues to satisfy the ECCS acceptance criteria as described in the UFSAR section 6.3.3. The manner for demonstrating the acceptability for this proposed change is that there is no adverse impact to the function of the HPSI system during the transfer of suction from the RWT to the containment sump. As previously discussed, the addition of a time delay to the opening circuit of valves 2CV-5649-1 and 2CV-5650-2 has no adverse impacts to the NPSH available to the suction of the HPSI (or CS) pumps and there is no air ingestion due to vortex formation at the RWT inlet. Therefore, it is concluded that the existing analysis in UFSAR section 6.3.3 for demonstrating the ECCS acceptance criteria remains valid and that the fuel cladding is not exceeded or altered by the proposed change.

In similar fashion, the containment boundary is not impacted as the function of the CS system as described in UFSAR section 6.2 and the containment boundary is not adversely impacted by the proposed change.

The closing circuit of valves 2CV-5649-1 and 2CV-5650-2 are not impacted by this change and the ability to close the valves should it be required to isolate excessive leakage in the recirculation piping is not adversely impacted.

Therefore, there is no design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered as a result of the proposed change.

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

BASIS:

The proposed change to add a time delay to the opening of the containment sump recirculation isolation valves 2CV-5649-1 and 2CV-5650-2 have been evaluated against the current design bases and safety analyses. The methods of analysis currently specified in UFSAR sections 6.2 and 6.3 have not been altered or departed from regarding the proposed change. This change has been evaluated and demonstrated to be bounded by the existing design bases and safety limits. Therefore, there is no departure from a method of evaluation described in the UFSAR 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 EN-LI-103.

ANO 50.59 Evaluation Number

2010-002

I. OVERVIEW / SIGNATURES[2]

Facility: Arkansas Nuclear One, Unit 2 **Evaluation # / Rev. #:** 10-002 / 0

Proposed Change / Document: EC-20413 / Add Zinc Injection Equipment to CVCS System for ANO Unit 2

Description of Change:

EC 20413 provides all necessary design documentation and justification to install and operate a Zinc Injection skid assembly in ANO-2.

The major objective of the zinc addition program at ANO-2 is to reduce the radiation dose rates associated with the primary system components. An ancillary benefit of zinc addition is that it is expected to inhibit the occurrence of PWSCC in Alloy 600 components.

All components added by EC 20413 are non-seismic, non-safety related, and non-quality related per EN-DC-167, System Assessment Safety Classification Worksheet.

The proposed design and installation of zinc injection components with connected/interfacing systems were evaluated and "screened out" in the PAD prepared for this EC. This 10CFR50.59 Evaluation addresses the addition of zinc to the primary system with affects on primary system components and fuel. Specifically, the following *potentially* adverse effects that screened-in to this evaluation are as follows:

- Boron dilution resulting from addition of the zinc solution from CVCS
- Crud deposits on fuel and the potential for Crud Induced Power Shift (CIPS)
- Crud deposits on fuel and the potential for Crud Induced Localized Corrosion (CILC)
- Zinc in sump post-LOCA and effects on sump strainers

The proposed change modifies the RCS chemistry control program at ANO-2 to allow for addition of a soluble zinc compound to the RCS during normal plant operation. Under the chemistry program modification proposed, a solution containing either natural or "depleted" zinc will be injected into the primary system via the CVCS.

Zinc is added to PWR reactor coolant in low concentrations to reduce plant radiation dose rates and reduce Primary Water Stress Corrosion Cracking (PWSCC) in susceptible plant materials. Zinc reacts with existing corrosion films on Reactor Coolant System (RCS) surfaces and makes the films more stable and resistant to further corrosion. Zinc will displace nickel, iron, cobalt, and other corrosion products from those films as it is incorporated into the films. These corrosion products may then deposit on the core as crud, increasing the risk of Crud Induced Power Shift (CIPS). These deposits, along with the change in chemistry environment experienced by the fuel, may increase the risk of enhanced corrosion of the fuel cladding, or Crud Induced Localized Corrosion (CILC).

These effects of injecting zinc into the primary system has been thoroughly evaluated by Westinghouse and summarized in the following two reports: Engineering Report CALC-A2-ME-2004-002, Revision 0, "Arkansas Nuclear One Unit 2 Primary System Zinc Injection Engineering Evaluation" (approved November 8, 2004) and Engineering Report CALC-ANO2-ME-10-00001, Rev. 0, "Westinghouse Fuel Risk Analysis for Zinc Addition at ANO-2" (added by EC 20413). These reports constitute the primary technical references for this Evaluation.

Westinghouse proposes the following zinc addition strategy: Zinc addition is to begin half way through the cycle with a target concentration of 5 ppb. The last half of Cycle 21 zinc addition is split, starting with 5 ppb concentration and increasing to 10 ppb for the last part. For Cycles 22 and 23 a full cycle of zinc addition with a 10 ppb target concentration starting at beginning of Cycle 22 and increasing to 20 ppb after half of cycle is completed with 20 ppb concentration being maintained for Cycle 23.

[2] Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

Nomenclature / Acronyms:

CT	Condensate Transfer System
RCS	Reactor Coolant System
CVCS	Chemical and Volume Control System
PWSCC	Primary Water Stress Corrosion Cracking
CIPS	Crud Induced Power Shift
BOA	Boron Induced Offset Anomaly
CILC	Crud Induced Localized Corrosion
T/H	Thermal / Hydraulic
ppb	Parts per Billion
lbm	Pounds (Mass)
NPSHA	Net Positive Suction Head Available

Is the validity of this Evaluation dependent on any other change? ☐ **Yes** ☒ **No**

If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation 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.

Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval? ☐ **Yes** ☒ **No**

Preparer: Christopher A. Davenport / See AS for signature / DP Engineering Ltd. / see AS for date
Name (print) / Signature / Company / Department / Date

Reviewer: Joseph A. Campbell / See AS for signature / DP Engineering Ltd. / see AS for date
Name (print) / Signature / Company / Department / Date

OSRC: Dale James / ORIGINAL SIGNED BY DALE E. JAMES / 07-15-10
Chairman's Name (print) / Signature / Date

OSRC 10-020
OSRC Meeting #

II. 50.59 EVALUATION

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 UFSAR?

☐ Yes
☒ No

BASIS:

The Moderator Dilution Incident (UFSAR 15.1.4), Loss of Coolant Accident (UFSAR 15.1.13), Steam Generator Tube Rupture (UFSAR 15.1.18), and Fuel Cladding Failure Combined with Steam Generator Leak (UFSAR 15.1.25) are the previously evaluated accidents / incidents that may be credibly affected by addition of zinc to the RCS. The impact on the frequency of occurrence of these accidents / incidents is addressed below.

Moderator Dilution Incident (UFSAR 15.1.4): The injected zinc acetate solution contains no boron and may have a dilution effect on the RCS boron concentration. An analysis has been performed by Westinghouse to confirm that the zinc injection rate has only a small effect on the RCS boron concentration. The analysis was based on a conservative flow rate of 3 gallons per hour. It is concluded that the zinc injection system has no adverse effect on plant operations or presents a challenge to the reactivity control of the primary system provided that controls (i.e., suspended / reduced zinc addition or increased boration rate) are implemented early in the cycle. Therefore, there is no measurable increase in the frequency of occurrence of a Moderator Dilution Incident as previously evaluated in the UFSAR.

Loss of Coolant Accident (UFSAR 15.1.13): The increase in the frequency of occurrence of a LOCA is evaluated based on the presence of zinc in the RCS. A LOCA, as it is evaluated in the UFSAR, results from a break in RCS piping up to and including a double ended rupture of the largest bore piping in the RCS. The addition of zinc has no deleterious effects on RCS materials, including piping, and may have a small beneficial effect on preventing PWSCC. Therefore, the addition of zinc is will not increase the frequency of occurrence of a LOCA as previously evaluated in the UFSAR.

Steam Generator Tube Rupture (UFSAR 15.1.18): The steam generator tube rupture as evaluated in the UFSAR involves a double ended break of a steam generator tube, resulting in a primary system leak to the secondary system in excess of charging capacity. Laboratory testing and PWR experience indicate that addition of zinc has a beneficial impact to Alloy 690 corrosion rates (as well as that of other RCS materials), thus zinc addition is expected to be beneficial to steam generator tube integrity. Therefore, the addition of zinc will not increase the frequency of occurrence of a steam generator tube rupture as previously evaluated in the UFSAR.

Fuel Cladding Failure Combined with Steam Generator Leak (UFSAR 15.1.25): This condition as evaluated in the UFSAR involves continuous operation with a 1 GPM primary-to-secondary leak with 1% failed fuel. The addition of zinc into the primary system has been evaluated for the effects on fuel cladding integrity and other RCS materials. As discussed under steam generator tube rupture above, zinc addition is expected to be beneficial to Alloy 690 steam generator tubes as well as other RCS materials. The effects of Crud Induced Localized Corrosion (CILC) on fuel cladding has been evaluated by Westinghouse with the conclusion that with the proposed zinc addition strategy, cladding damage from build-up of crud remains low. Therefore, the addition of zinc will not increase the frequency of occurrence of fuel cladding failure combined with steam generator tube leakage as previously evaluated in the UFSAR.

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 UFSAR?

☐ Yes
☒ No

BASIS:

The containment sump strainers are evaluated to address the effects of zinc in the sump post accident. A malfunction such as strainer clogging due to chemical effects of zinc addition is addressed. The effects of zinc on primary system materials (including fuel) are addressed. A malfunction due to corrosion of RCS materials, fuel cladding and changes in crud deposition that could potentially result in localized cladding corrosion is addressed.

In the event of a LOCA, reactor coolant with concentrations of zinc will collect in the Containment sump. The following Westinghouse evaluations conclude that zinc will not increase the likelihood of occurrence of a malfunction of the ECCS including the sump screen components previously evaluated in the UFSAR.

> Debris blockage of containment sump screens can impede or prevent emergency core cooling system (ECCS) and containment spray system operations. Of additional concern is the compound effect of debris, fiber and precipitating materials on sump screen performance. For a conservatively high initial RCS zinc concentration of 100 ppb, the resulting sump zinc concentration was calculated to be 2.7×10^{-6} M in the event of a LOCA. This concentration is sufficiently low that it has been determined there is no significant impact related to containment sump screen performance as a result of adding zinc to the reactor coolant system.
>
> To determine the effect of zinc on the sump pH, the effect of zinc on the RCS was first evaluated. The net effect on the at-temperature pH of the RCS was calculated to vary from 0.0 to 0.02 pH units. This effect is judged negligible. The effect on the sump pH is even smaller, because the RCS volume is a small fraction of the borated water volume entering the sump.

The following evaluations conclude that the zinc exposure to fuel and RCS materials will not increase the likelihood of occurrence of a malfunction of these components as previously evaluated in the UFSAR.

> A fuel assessment concludes that ANO-2 Cycle 21 fuel will meet all fuel rod corrosion and design limits assuming zinc addition. Fuel examinations will be required. For future cycles, ZIRLO clad fuel will meet all fuel rod corrosion and design limits. OPTIN Zircaloy-4 clad fuel will require cycle-specific zinc addition evaluations, but it is expected to support zinc addition strategies. Strategies have been developed to minimize any increase in CIPS risk for the addition of zinc in ANO-2 Cycle 21. These include delaying zinc addition to approximately eight months into the cycle and starting zinc addition at lower concentrations than have already been used in zinc addition plants that did not experience significant levels of CIPS.
>
> The current understanding of the beneficial effects of zinc on general corrosion and stress corrosion cracking of Alloy 600 indicates that they are the results of modification of the oxide corrosion films that develop on primary system materials in primary coolant containing zinc. Zinc causes the development of thinner oxide films and also modifies the structure and morphology of these spinel corrosion films, leading to the preferential release of nickel and cobalt by the substitution of zinc for these elements in the spinel lattice. The general aspects and benefits of zinc addition are well established and accepted. Zinc has a positive effect by lowering radiation fields. Zinc addition also inhibits the initiation of PWSCC in primary system materials.

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

☐ Yes
☒ No

BASIS:

Uncontrolled Boron Dilution Incident (UFSAR 15.1.4): The UFSAR conclusions for the moderator dilution event for multiple plant conditions are that a sustained uncontrolled boron dilution event is unlikely in view of the administrative procedures and system design and in view of numerous indications available to the operator. However, should the postulated events discussed above occur, the maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the dilution and take corrective action before significant shutdown margin is lost.

Due to the zinc injections strategy for ANO-2, only Dilution During Critical Operation (UFSAR 15.1.4.2.6) is considered here. The injected zinc acetate solution contains no boron and may have a dilution effect on the RCS boron concentration. An analysis has been performed to confirm that the zinc injection rate has only a small effect on the RCS boron concentration. The analysis was based on a conservative flow rate of 3 gallons per hour. It is concluded that the existing evaluations and protection methods are bounding and therefore the proposed zinc injection system has no adverse effect on plant operations or presents a challenge to the reactivity control of the primary system provided that controls (i.e., suspended / reduced zinc injection or increased boration rate) are implemented early in the cycle.

Therefore, as the rate and quantity of zinc addition is slow the conclusions reached in the UFSAR for this event remain valid and there will be no more than a minimal increase in the consequences of a boron dilution incident previously evaluated in the UFSAR.

Steam Generator Tube Rupture (UFSAR 15.1.18) and Fuel Cladding Failure Combined with Steam Generator Leak (UFSAR 15.1.25): For this accident, the addition of zinc to the RCS is evaluated for any potential increase in the radiological consequences of a steam generator tube rupture as evaluated in the UFSAR. Multiple assessments of the radiological aspects of zinc addition for a variety of plants have been previously made by Westinghouse. These assessments were based upon the existing experience relative to the radiological aspects of zinc addition. Evaluations have been required since radioactive ^{65}Zn is formed by the activation of ^{64}Zn in natural zinc. Based on Westinghouse experience, the amounts of ^{58}Co and ^{60}Co in the coolant are expected to increase until conditioning of the primary system corrosion product deposits is complete.

Therefore, the addition of natural or depleted zinc to ANO-2 will not have a significant effect on the radiological operation of the plant. This assessment is made with respect to mixed bed demineralizer dose rates, Technical Specification limits for radionuclides in the coolant, and equipment qualification dose.

The radiological consequences of the evaluated steam generator tube rupture and fuel cladding failure combined with steam generator leakage are not increased in that the UFSAR analyses assume operation with 1% fuel failure prior to/during the event. This assumption continues to "bound" the analyses conclusions with respect to radiological releases.

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 UFSAR?

☐ Yes
☒ No

BASIS:

The addition of zinc into the RCS does not introduce a change in the consequences of a malfunction. 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 addition. Based on Westinghouse's fuel and sump screen evaluations and current industry experience, zinc will not result in fuel failures beyond that already analyzed in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the UFSAR? ☐ 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 UFSAR. The zinc affects on fuel cladding and RCS wetted materials has been evaluated to rule out potential for gross failures of cladding or RCS pressure boundaries beyond those evaluated in the UFSAR.

The primary cladding failure mechanism examined is CILC. CILC results from thick local crud deposits and can lead to cladding failure. If local crud deposits become unexpectedly thick, then the liquid coolant may no longer be able to penetrate the crud and flow to the clad surface. In this unlikely case, only a steam layer is available to cool the clad surface resulting in elevation of clad temperatures, which can lead to accelerated corrosion and cladding failure. Westinghouse analyses for the first cycle of zinc addition, results in recommendations for timing and RCS chemistry that should limit the potential for CILC. Westinghouse analyses predict low risk of crud dry-out, and though some local crud thicknesses may increase with zinc addition, the overall CILC risk for ANO-2 remains low.

The general aspects and benefits of zinc addition on primary system materials are well established by both Westinghouse and industry experience, including inhibiting the initiation of PWSCC in primary system materials.

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 UFSAR? ☐ Yes ☒ No

BASIS:

With the small, controlled addition of zinc, crud buildup and increased cladding oxidation will be acceptable and insignificant. The expected increase is well within the experience base at other plants with higher fuel duty. Furthermore, the Westinghouse evaluation of the effects on containment sump screens concluded that there is no significant impact to sump screen performance from zinc addition and a negligible change to sump pH.

The results from failure for fuel cladding from crud deposition remain the same with release of radionuclides to the RCS after zinc addition. The results from clogging of the sump screens that is reduction in ECCS NPSHA and/or flow remain the same after zinc addition. The chemical and oxidation effects from zinc in the prescribed concentrations introduce no new results from the failures of these SSCs. Therefore, 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.

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

BASIS:

From a fuel integrity perspective, with a small increase in cladding oxidation and fuel cladding temperatures being the only effects of the zinc addition, adequate margins will remain to the existing analysis limits. The margin of safety for the fuel was addressed by Westinghouse against the applicable design criteria and the impact of the zinc addition of the fuel evaluation.

The Westinghouse fuel evaluation and how it affects the UFSAR is summarized below.

> Advanced BOA models have been generated for ANO-2 Cycles 21-23. The thermal/hydraulic model results show that the ANO-2 boiling duty is bounded by other plants such as Vogtle and Byron 2 that have added zinc. A CIPS/CILC analysis consistent with the requirements of EPRI "Fuel Reliability Guidelines, PWR Fuel Cladding Crud and Corrosion" was performed for upcoming ANO-2 cycles.

Though addressed in Westinghouse analyses, the CIPS/CILC are not specifically addressed in the UFSAR (or other licensing basis documents).

Fuel examinations to date have not shown an increase in clad corrosion resulting from zinc addition in fuel with ZIRLOTM clad. Therefore standard corrosion modeling methods were used. Results indicated that margin remains to the clad corrosion limits.

Though addressed in Westinghouse analyses, the clad corrosion limits are established by the fuel supplier and are not specifically stated in the UFSAR (or other licensing basis documents). Therefore, such a limit is not exceeded or altered.

BOA predictions for Cycles 21-23 result in low CIPS risk based on the proposed zinc addition strategy as described before and no reactor trips modeled. Should a reactor trip occur, CIPS risk could be potentially elevated due to additional corrosion product release from Control Rod Drive Mechanism. With the worst case in assuming a reactor trip occurs at 30 days into the cycle, ANO-2 Cycle 21 Fuel Rod Corrosion and Zinc Injection Evaluation, November, 2009, indicates an increased maximum boron buildup to 0.292 lbm for Cycle 21, which is slightly above the low CIPS risk threshold of 0.28 lbm.

The Advanced BOA CILC results also predict low risk of crud dry-out. Local crud thicknesses increase with the proposed zinc addition strategy, but overall CILC risk remains low.

The zinc addition strategy assumed for ANO-2 begins in Cycle 21 after the first half of the cycle is complete. A target zinc concentration of 5 ppb is assumed for the remainder of the cycle, increasing to 10 ppb about half way through second half of cycle. Subsequent cycles assumed a target zinc concentration of 10 ppb at beginning of Cycle 22 and increasing to 20 ppb after half of cycle has completed and 20 ppb concentration maintained for entire Cycle 23.

The degradation of leaking fuel observed in zinc addition plants was very similar to the plants and cycles without zinc. The degradation of the leaking rods depended mainly on the type of defect, the duty of the leaking rod, and the time of exposure to the coolant. The review concluded that there is no concern of additional degradation of the leaking fuel due to zinc injection.

The RCS pressure boundary is not degraded by the addition of low concentrations of zinc to the primary water. The current understanding of the beneficial effects of zinc on general corrosion and stress corrosion cracking of Alloy 600 indicates that they are the results of modification of the oxide corrosion films that develop on primary system materials in primary coolant containing zinc. Zinc causes the development of thinner oxide films and also modifies the structure and morphology of these films, leading to the preferential release of nickel and cobalt by the substitution of zinc for these elements in the oxide films. The general aspects and benefits of zinc addition are well established and accepted. Zinc has a positive effect by inhibiting the initiation of PWSCC in primary system materials.

Zinc addition will have no effect on containment integrity (including penetration points) because the liner and isolation points will not be exposed to zinc in other than trace concentrations. At these concentrations, zinc will have no effect. Zinc in the RCS is incorporated into oxide layers of the RCS, and zinc in the reactor coolant exists as divalent cations and not neutral zinc atoms in solution. During a post-accident scenario, zinc residuals cannot release hydrogen thus the existing hydrogen generation within the containment is not adversely impacted.

For the above reasons, zinc addition will not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered.

8. Result in a departure from a method of evaluation described in the UFSAR 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 zinc addition was 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 analyses. 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.

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 EN-LI-103.

ANO 50.59 Evaluation Number

2010-003

I. OVERVIEW / SIGNATURES[3]

Facility: ANO-2 **Evaluation # / Rev. #:** 10-003

Proposed Change / Document:

A new methodology is added to the ANO-2 Cycle 21 Core Operating Limits Report (COLR):

"Addendum 1 to Topical Report WCAP-16500-P, Supplement 1, Revision 1, "Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF)"

Description of Change:

A PAD was performed which addressed changes to the Core Operating Limits Report (COLR) that will align the COLR with the recently approved Amendment 290 to the ANO-2 Technical Specifications (TSs). These changes met the screening criteria of the PAD and are not addressed in this 10 CFR 50.59 Evaluation.

As COLR changes were being prepared to address the above amendment, the NRC approved on July 1, 2010, an additional Westinghouse methodology applicable to ANO-2. That document addresses improved DNBR penalty factors that will be applied in the future. This new approved methodology is being incorporated into the COLR at this time by adding the new approved methodology to the renumbered methodologies noted above. While it is acceptable to adopt this new methodology without further NRC approval, NEI 96-07, Rev. 1, Guidelines for 10 CFR 50.59 Implementation, requires the justification discussion for new methodologies to be included in a 10 CFR 50.59 Evaluation. In accordance with Section II of this form, only Question 8 is answered, relevant to changes in methodologies.

Is the validity of this Evaluation dependent on any other change? ☐ **Yes** ☒ **No**

If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation 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.

Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval? ☐ **Yes** ☒ **No**

Preparer: James H. Willoughby / See LAR 2010-0190 / EOI / Fuels & Analysis / 08-12-10
Name (print) / Signature / Company / Department / Date

Reviewer: David Bice / See LAR 2010-0190 / EOI / Licensing / 08-12-10
Name (print) / Signature / Company / Department / Date

OSRC: Dale James / ORIGINAL SIGNED BY DALE E. JAMES / 08-19-10
Chairman's Name (print) / Signature / Date

OSRC-10-025
OSRC Meeting #

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[3] Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

II. 50.59 EVALUATION

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 UFSAR?
   ☐ Yes
   ☐ No

   BASIS:

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 UFSAR?
   ☐ Yes
   ☐ No

   BASIS:

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

   BASIS:

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 UFSAR?
   ☐ Yes
   ☐ No

   BASIS:

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

   BASIS:

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 UFSAR?
   ☐ Yes
   ☐ No

   BASIS:

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

   BASIS:

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

   BASIS:

   The NRC approved on July 1, 2010, an additional Westinghouse methodology applicable to ANO-2 (Reference 1). That document addresses improved Departure from Nucleate Boiling (DNBR) penalty factors that will be applied in the future. In accordance with NEI 96-07, Rev. 1, Guidelines for 10 CFR 50.59 Implementation, NRC approved methodologies may be adopted for use by a facility provided the method is applicable to the facility and any conditions stipulated in the NRC Safety Evaluation Report (SER) for the given methodology have been met by the facility.

Facilities that adopted the use of Next Generation Fuel were initially imposed with a 6% DNBR penalty factors to provide additional safety margin during the first few cycles of operation with NGF. ANO-2 continues to carry this penalty. The newly approved methodology (Reference 1) removes this penalty since several facilities now have one or more cycles of operation using NGF. The NRC had been collecting data from these facilities over time and has concluded that the 6% penalty is no longer required. Therefore, the new methodology is adopted into the ANO-2 COLR and the 6% penalty removed at a later date as designated by Reactor Engineering.

The SER contained no limits or conditions for adoption of the methodology or removal of the 6% penalty.

Based on the above and the guidance provided in NEI 96-07, this methodology may be adopted into the ANO-2 COLR without further NRC approval.

References:

1. Letter, Blount (NRC) to Gresham (Westinghouse), “”Final Safety Evaluation for Westinghouse Electric Company Addendum 1 to Topical Report WCAP-16500-P, Supplement 1, Revision 1, “Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF),” (TAC No. ME3583),”” July 1, 2010.

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 EN-LI-103.