UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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September 16, 2016

Mr. Brian D. Boles
Site Vice President
FirstEnergy Nuclear Operating Co.
Davis-Besse Nuclear Power Station
5501 N. State Rte. 2, Mail Stop A–DB–3080
Oak Harbor, OH 43449–9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION – CLOSURE OF UNRESOLVED ITEM 05000346/2013010-01 – INSPECTION REPORT 05000346/2016010

Dear Mr. Boles:

On September 7, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed the inspection of Unresolved Item 05000346/2013010-01, "Title 10 of the *Code of Federal Regulations* 50.59 Safety Evaluation Not Performed for Calculated Component Damping," identified during the replacement steam generator inspection at your Davis-Besse Nuclear Power Station [reference Inspection Report 05000346/2013010, dated July 22, 2014, (Agencywide Documents Access and Management System (ADAMS) Accession Number ML14204A317)]. The enclosed report documents the results of this inspection, which were discussed on September 7, 2016, with you and other members of your staff.

Based on the results of this inspection, one NRC-identified finding of very-low safety significance was identified. The finding involved a violation of NRC requirements. However, because of the very-low safety significance, and because the issue was entered into your Corrective Action Program, the NRC is treating the issue as a Non-Cited Violation (NCV), consistent with Section 2.3.2 of the Enforcement Policy. This NCV is described in the subject inspection report.

If you contest the violation or significance of this NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555–0001, with copies to: (1) the Regional Administrator, Region III; (2) the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001; and (3) the NRC Resident Inspector at the Davis-Besse Nuclear Power Station.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, "Public Inspections, Exemptions, Requests for Withholding," of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC's Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

David E. Hills, Chief
Engineering Branch 1
Division of Reactor Safety

Docket No. 50–346
License No. NPF–3

Enclosure:
IR 05000346/2016010

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No.:	50-346
License No.:	NPF-3
Report No.:	05000346/2016010
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Davis-Besse Nuclear Power Station
Location:	Oak Harbor, OH
Dates:	February 15, 2016 – September 7, 2016
Inspector:	J. Neurauter, Senior Reactor Engineer
Approved by:	David E. Hills, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY

Inspection Report 05000346/2016010; 02/15/2016 – 09/07/2016; Davis-Besse Nuclear Power Station; Closure of Unresolved Item 05000346/2013010-01.

This report covers inspection performed to close Unresolved Item 05000346/2013010-01 identified during the 2014 replacement steam generator inspection. The inspection was conducted by a Region III based engineering inspector. Based on the results of this inspection, one finding of very-low safety significance (Severity Level IV – Green) was identified by the inspector. The finding was considered a Non-Cited Violation (NCV) of U.S. Nuclear Regulatory Commission (NRC) regulations. The significance of inspection findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas," dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated February 4, 2015. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG–1649, "Reactor Oversight Process," dated July 2016.

NRC-Identified and Self-Revealed Findings

Cornerstone: Barrier Integrity

Severity Level IV-Green. A finding of very-low safety significance and an associated NCV of Title 10 of the *Code of Federal Regulations* (CFR), Part 50.59(b)(1), "Changes, Tests, and Experiments," (effective January 1, 1991) was identified by the inspector for the licensee's failure to maintain records that included a written safety evaluation which provided the bases for determining that the change to seismic licensing basis damping in calculations to support removal of snubbers under modification 90-0079 did not involve an unreviewed safety question. Specifically, licensee safety evaluation SE91-0046 did not provide a suitable basis for concluding that there was no increase in the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report, in that it did not address how the basis for the NRC's-approval of the seismic design of the reactor coolant system continued to be met with respect to the steam generator slider support (Lubrite plate) damping. In particular, a May 31, 1983, NRC Safety Evaluation Report approved the licensee's use of 0.15g safe shutdown earthquake ground acceleration in its seismic analysis for reactor coolant system design, in part, because "there is sufficient conservatism and margin in the piping systems components and supports at Davis-Besse Unit 1 to ensure safe shutdown and continued shutdown heat removal in the event of a safe shutdown earthquake having a ground acceleration of 0.20g." The licensee subsequently adopted a significantly higher damping value for the steam generator slider support while maintaining a 0.15g acceleration for the design without addressing how "sufficient conservatism and margin" otherwise continued to be met. The licensee entered this issue into its corrective action program.

The inspector determined that the licensee's failure to provide in its 10 CFR 50.59 evaluation, SE91-0046, a suitable basis for the determination that the use of damping higher than established in the seismic licensing basis for the reactor coolant system, specifically the steam generator slider support, was not an unreviewed safety question was a performance deficiency. The issue of concern was determined to be more than minor because the performance deficiency impacted the Barrier Integrity cornerstone

objective to provide reasonable assurance that physical design barriers (reactor coolant system) protect the public from radionuclide releases caused by accidents or events and the design control attribute to maintain functionality of the reactor coolant system. The inspector evaluated the underlying technical issue using IMC 0609, “The Significance Determination Process for Findings at Power,” Appendix A, Exhibit 1, “Initiating Events Screening Questions.” The inspector answered “No” to all the questions in Exhibit 1. In particular, because the reactor coolant system remained operable (capable of performing its safety function during a seismic event), the finding was determined to have very-low safety significance (Green) corresponding to a Severity Level IV violation per Example 6.1.d.2 of the NRC Enforcement Policy. The inspector did not identify a cross-cutting aspect associated with the finding because the finding was not representative of current performance. (Section 4OA5.1).

Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

4. OTHER ACTIVITIES

Cornerstone: Barrier Integrity

4OA5 Other Activities

.1 (Closed) Unresolved Item (URI) 05000346/2013010-01; "10 CFR 50.59 Safety Evaluation Not Performed for Revised Component Damping"

a. Inspection Scope

At the end of the replacement steam generators inspection in 2014, the inspector required additional information to determine if the licensee's Updated Safety Analysis Report (USAR) changes and associated Title 10 of the *Code of Federal Regulations* (CFR) 50.59 evaluations provided appropriate rationale to support a conclusion that prior U.S. Nuclear Regulatory Commission (NRC)-approval was not required for an increase to seismic design and licensing basis damping, refer to Inspection Report 05000346/2013010, dated July 22, 2014, (Agencywide Documents Access and Management System (ADAMS) Accession Number ML14204A317).

The Davis-Besse steam generator slider support (Lubrite plate) provides a bearing surface upon which a steam generator rests for vertical support and allows horizontal movement. During review of the replacement of the steam generators, the inspector identified that the licensee used a percentage of critical damping value for the steam generator slider support that exceeded the Davis-Besse Nuclear Power Station (DBNPS) USAR damping applicable to the seismic analysis for the DB-1 Reactor Coolant System (RCS) 1/2-Loop Model coupled system. Specifically, licensee calculation 32-9191681-002, "DB-1 RCS 1/2-Loop Model and Loading Analysis," Revision 2, that evaluated the effects of the replacement steam generator, utilized friction properties associated with the steam generator slider support to calculate an effective component damping factor and modified the seismic response spectra for operating basis earthquake (OBE) and safe shutdown earthquake (SSE). Damping affects energy loss (for example, friction and joint slippage dissipate energy) within the structure and therefore limits structural motion during a seismic event. Hence, if the assumed damping value is higher, the calculated piping and component stresses will be lower for the postulated seismic event.

The licensee's 10 CFR 50.59, Screen 12-03404, did not address the fact that this effective damping was higher (less conservative) than defined in the USAR. In discussions with the inspector, the licensee justified not evaluating this change to the USAR damping value in the screening because calculation 32-9191681-002 used damping that was conservative (lower) than damping used in the design basis analysis for the DB-1 RCS 1/2-Loop Model and Loading Analysis that existed prior to steam generator replacement, calculation 32-1177088-06, "TE 1/2-Loop Model Reanalysis," Revision 6.

The licensee had previously modified USAR damping for the steam generator slider support in the RCS 1/2-Loop Model in calculation 32-1177088-00 (Revision 0) in 1990, as part of plant modification 90-0079 that removed snubber seismic restraints at the base of the original steam generators. The supporting 10 CFR 50.59 safety evaluation (SE), SE 91-0046, Revision 1, did not specifically address this increase to the USAR damping specified for the RCS 1/2-Loop Model coupled system.

During discussions with the inspector, licensee staff claimed that it was appropriate to use a higher effective component damping factor in its analysis that credited the additional friction associated with the steam generator slider support because the friction was not considered in existing damping values. However, the inspector noted that friction is a fundamental element of damping and already conservatively considered to the extent deemed appropriate in the determination of damping values in the plant's licensing basis.

During the current inspection, the inspector reviewed NRC Safety Evaluation Reports (SERs) and the Davis-Besse USAR associated with Davis-Besse seismic licensing basis damping, seismic damping values for the DB-1 RCS 1/2-Loop Model coupled system, USAR seismic damping values used in design calculations associated with the DB-1 RCS 1/2-Loop Model coupled system, the 10 CFR 50.59 rule in effect in 1991, and the licensee's 10 CFR 50.59 evaluation, SE 91-0046, that supported removal of snubber seismic restraints at the base of the original steam generators. The inspector also held discussions with licensee staff to ascertain additional licensee evaluation, rationale, and conclusions regarding the concern, and consulted with staff in the NRC's Office of Nuclear Reactor Regulation. Specifically, the inspector reviewed this information to more fully ascertain the associated licensing basis, determine how that was addressed in the licensee's 10 CFR 50.59 screenings and evaluations, and evaluate whether 10 CFR 50.59 requirements in effect at the time were appropriately met.

b. Findings

(1) 10 CFR 50.59 Evaluation Failed to Properly Evaluate Change to Seismic Licensing Basis

Introduction: A finding of very-low safety significance (Green) and associated Severity Level IV Non-Cited Violation (NCV) of 10 CFR Part 50.59, "Changes, Tests, and Experiments," was identified by the inspector for the licensee's failure to maintain records that included a written safety evaluation which provided the bases for determining that the change to seismic licensing basis damping in calculations to support removal of snubbers under modification 90-0079 did not involve an unreviewed safety question (USQ). Specifically, licensee safety evaluation SE 91-0046 did not provide a suitable basis for concluding that there was no increase in the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report, in that it did not address how the basis for the NRC's-approval of the seismic design of the reactor coolant system continued to be met with respect to the steam generator slider support (Lubrite plate) damping.

Description:

Plant Specific Seismic Analysis Licensing Basis for the Reactor Coolant System Incorporated Specific Expectations for Conservatism and Margin

The Davis-Besse Unit 1 facility was designed prior to the issuance of Appendix A to 10 CFR Part 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," and Regulatory Guide (RG) 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants." In particular, with respect to this specific issue, the Babcock & Wilcox half loop model analysis that included hydraulic snubbers at the base of each steam generator for Davis-Besse was issued in 1972. Davis-Besse structures and components were designed for an SSE acceleration of 0.15g at the foundation level. As documented in Section 18 of NUREG-0136, "Safety Evaluation Report Related to Operation of Davis-Besse Nuclear Power Station Unit 1," Supplement 1, dated April 1977, an NRC Advisory Committee on Reactor Safeguards (ACRS) believed that an acceleration of 0.2g for an SSE would be more appropriate for the Davis-Besse site. As documented in Section 3.7 of this supplement, NRC staff agreed with the ACRS, and in Section 2.5, NRC staff indicated that Toledo Edison Company was required to re-evaluate portions of the Davis-Besse plant for an SSE acceleration of 0.2g applied at the foundation of the plant. Specifically, NRC staff was to review in detail the plant systems needed to accomplish a safe shutdown of the reactor and continued shutdown heat removal for an SSE acceleration of 0.2g and that RG 1.60 guidance should be applied at the foundation level of the facility.

The Davis-Besse operating license was issued on April 22, 1977, and the re-analysis submittal was required via license condition 2.C.(3)(r):

> "Toledo Edison Company shall submit a seismic re-analysis and evaluation to the Commission for its review and approval in sufficient time to obtain Commission approval of the adequacy of the plant systems needed to accomplish safe shutdown of the reactor and continued heat removal prior to startup following the first regularly scheduled refueling outage. In performing the re-analysis a safe shutdown earthquake acceleration of 0.20g shall be applied at the foundation level of the plant and the response spectra shall be used as specified in Regulatory Guide 1.60, 'Design Response Spectra for Seismic Design of Nuclear Power Plants'. The seismic reanalysis and evaluation shall be conducted in accordance with guidelines to be specified by the Commission."

Based on licensee submitted re-analysis, by letter dated August 27, 1980, (ADAMS Accession No. ML02120322), the Commission issued License Amendment No. 30 to Facility Operating License No. NPF-3 for Davis-Besse. Specifically, the amendment deleted License Condition 2.C.(3)(r) regarding submittal of a seismic reanalysis. In the enclosed safety evaluation for License Amendment No. 30, NRC staff concluded, "Our preliminary review indicates that the licensee has demonstrated that there is sufficient conservatism and margin at DB-1 to ensure safe shutdown and continued shutdown heat removal in the event of an SSE having a ground acceleration of 0.20g. On this basis, we conclude that operation of the plant following the completion of the first refueling outage may continue pending completion of our review of the licensee's reanalysis."

The NRC staff subsequently documented its review of Davis-Besse plant systems needed to accomplish safe shutdown and continued heat removal for RG 1.60 response spectra based on an SSE acceleration of 0.2g applied at the foundation level of the facility in a May 31, 1983, SER. The inspector reviewed the NRC SER and noted:

> The limited seismic reanalysis focused on 11 piping systems (or portions thereof) including the reactor coolant system (for pressure boundary), eight ventilation systems, and selected mechanical equipment, electric equipment, and instrumentation.
>
> As an example, for the evaluation of piping systems, two earthquakes were considered:
>
> 1. An SSE with a ground acceleration of 0.15g and 1/2 percent damping (the DBNPS design earthquake); and
>
> 2. An SSE with a ground acceleration of 0.20g and 2 percent of critical damping aligning more closely to RG 1.61 "Damping Values for Seismic Design of Nuclear Power Plants" (the revised earthquake).
>
> For the piping systems evaluated in the seismic reanalysis sample, the SER noted that "the stresses with the 0.20g acceleration and 2 percent of critical damping are lower than the values with 0.15g and 1/2 percent damping."
>
> After detailing similar reanalysis for the other examples, the SER concluded that "the licensee has demonstrated that there is sufficient conservatism and margin in the piping systems components and supports at Davis-Besse Unit 1 to ensure safe shutdown and continued shutdown heat removal in the event of an SSE having a ground acceleration of 0.20g."

In summary, although the 0.15g acceleration evaluated in the licensee's original analysis was non-conservative to the 0.20g acceleration desired for evaluation, the more conservative (lower) damping values in the original analysis (compared to damping values that were aligned more closely to RG 1.61) resulted in higher calculated stresses when compared to stress acceptance criteria. Hence, the overall approach in the original analysis (0.15g acceleration with much lower damping values than specified in RG 1.61) was determined to be conservative to 0.2g acceleration using RG 1.61 damping values. This conservatism, combined with margin in piping system components and supports, served as the basis for the NRC's-approval of the licensee's seismic analysis that considered the lower 0.15g acceleration earthquake. Subsequent use of less conservative damping values would necessitate an evaluation to demonstrate "sufficient conservatism and margin in the piping systems components and supports at Davis-Besse Unit 1 to ensure safe shutdown and continued shutdown heat removal in the event of an SSE having a ground acceleration of 0.20g."

Hence, the licensee maintained an SSE with a ground acceleration of 0.15g, but with conservative damping as the DBNPS design earthquake. As a part of the July 1983 USAR update, the licensee revised the USAR to address the seismic re-evaluation and the NRC SER in Section 2C.3.5 of Appendix 2C. As part of the November 2000 USAR update, the licensee further revised Section 2C.3.5 of Appendix 2C to document that DBNPS continued to use seismic design parameters described in Section 2C.3.4, 0.08g for the maximum probable earthquake (OBE) and

0.15g for the maximum possible earthquake (SSE). In addition, USAR Section 3.7.1.3 defined seismic damping values to which the plant has been licensed for analyzing a structure, system, or component (SSC): Table 3.7-1 (above the dashed line) and Section 3.7.2.14.

Subsequent to the May 31, 1983, NRC SER, the inspector noted an NRC-approval that allowed the licensee to apply damping that was higher than RG 1.61 damping values at Davis-Besse. Specifically, in Toledo Edison letter dated July 2, 1985, (Serial No. 1163), the licensee requested NRC-approval to use American Society of Mechanical Engineers (ASME) Code Case (CC) N-411, “Alternative Damping Values for Seismic Analysis of Classes 1, 2, and 3 Piping, Section III, Division 1,” dated September 17, 1984, for response spectrum seismic analyses, reconciliation work, modifications, and support optimization in lieu of 0.5 percent damping specified in subsection 3.7.1.3 and Table 3.7-1 of the USAR. In a letter to the licensee dated August 26, 1985, the NRC conditionally approved this use of CC N-411.

The use of higher damping values in CC N-411 was approved via Code relief provisions in the regulations, and hence the change was not subject to the license amendment review and approval process. In particular though, the requests and approvals did not address the Davis-Besse unique, plant specific licensing basis that justified designing to the lower 0.15g SSE ground acceleration, specifically “that there is sufficient conservatism and margin in the piping systems components and supports at Davis-Besse Unit 1 to ensure safe shutdown and continued shutdown heat removal in the event of an SSE having a ground acceleration of 0.20g.” By itself, that would not be an issue of concern, in that the approval reflected technical soundness of the CC damping values and use of those values would not necessarily be inappropriate provided the licensee addressed in a 10 CFR 50.59 evaluation how the “sufficient conservatism and margin” in its licensing basis otherwise continued to be met when CC N-411 values were used.

<u>Significantly Less Conservative Damping Value for the Steam Generator Slider Support</u>

The inspector reviewed design calculations for seismic damping applied in the RCS ½-Loop Model coupled system analyses:

- The original design analysis, “Stress Analysis for Toledo Edison Company (NSS-14) Primary Code Piping,” dated April 19, 1972, used “a conservative damping value of 1/2 percent for all components” consistent with USAR Section 3.7.2.14.

- The design analysis supporting the 1990 removal of snubbers at the base of the original steam generators, calculation 32-1177088-00, modified original seismic damping to 16.3 percent of critical for steam generator response: 12.6 percent due to steam generator / Lubrite plate friction; 3.7 percent due to structural damping.

- The design analysis supporting the replacement steam generators, calculation 32-9191681-002, modified the following seismic damping to evaluate the effect of the replacement steam generator:

- Fourteen percent of critical for steam generator response (OBE): 10.4 percent due to steam generator / Lubrite plate friction; 3.7 percent due to structural damping.

- Ten percent of critical for steam generator response (SSE): 5.9 percent due to steam generator / Lubrite plate friction; 4.3 percent due to structural damping.

- The above damping values used for the steam generator slider support in both the 1990 removal of snubbers and the subsequent replacement of the steam generators exceeded even the CC N-411 allowed damping values.

The licensee changed the design basis earthquake for the RCS from response spectra based on an SSE ground acceleration 0.15g at foundation level with ½ percent damping to response spectra based on considerably less conservative damping. Therefore, the licensee decreased the original seismic design loads in calculation 32-1177088-00 and calculation 32-9191681-002, which resulted in reduced calculated stresses.

The inspector concluded that the licensee's approach, maintaining 0.15g ground acceleration but adopting a damping value substantially greater than that specified in RG 1.61 for the steam generator slider support was not conservative to an analysis for an SSE ground acceleration of 0.20g at foundation level with RG 1.61 damping. In addition, the licensee did not perform an evaluation to demonstrate how the Davis-Besse unique, plant specific licensing basis, that justified designing to the lower 0.15g SSE ground acceleration, continued to be met. In particular, the licensee did not demonstrate that there continued to be "sufficient conservatism and margin in the piping systems components and supports at Davis-Besse Unit 1 to ensure safe shutdown and continued shutdown heat removal in the event of an SSE having a ground acceleration of 0.20g."

<u>Application of 10 CFR 50.59-1991 Rule to the Change to the Reactor Coolant System Seismic Analysis With Respect to the Steam Generator Slider Support</u>

As discussed previously in this inspection report, the initial change to RCS seismic licensing basis damping occurred in calculation 32-1177088-00 that supported plant modification 90-0079 that removed the snubber seismic restraints at the base of the original steam generators. The inspector reviewed the adequacy of the licensee's 10 CFR 50.59 evaluation Safety Evaluation (SE) 91-0046 that supported modification 90-0079. The licensee's procedure, NG-EN-00304, "Safety Review and Evaluation," effective January 18, 1990, provided instructions for the initiation, preparation, review, and approval of safety reviews and evaluations, as required by 10 CFR 50.59. Procedure NG-EN-00304 referenced Nuclear Safety Analysis Center (NSAC)-125, "Guidelines for 10 CFR 50.59 Safety Evaluations," dated June 1989. In SECY-97-035, "Proposed Regulatory Guidance Related to Implementation of 10 CFR 50.59 (Changes, Tests and Experiments)," dated February 12, 1997, NRC documented that although the NRC staff had not endorsed NSAC-125 industry guidance and NSAC-125 guidance was not required for any licensee, NSAC-125 had given the nuclear power industry a reasonable foundation to establish a process that will, in most instances, produce effective evaluations related to changes to plant design or procedures. Therefore, the inspector focused on the version of the 10 CFR 50.59 rule in effect at the time the licensee's evaluation was generated with insights provided in NSAC-125.

Title 10 CFR 50.59(a)(1) (effective January 1, 1991) allows the licensee to make changes in the facility as described in the safety analysis report, without prior Commission approval, unless the change involves a change in the Technical Specifications or a USQ. Also, 10 CFR 50.59(a)(2) specifies that a proposed change shall be deemed to involve a USQ if one or more specific criteria is met, one of which is "if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased."

Further, NSAC-125, Section 3.3 states that: "the term "malfunctions" of equipment important to safety refers to the failure of SSCs to perform the safety functions described in the safety analysis report (SAR). The occurrence of malfunctions could be caused by failure to comply with design considerations, such as code requirements, pipe whip, jet impingement, seismic, fire, flooding, tornado, missiles, etc." That section also states that: "the SER is a source of information that may be useful in determining acceptance criteria for provisions in the SAR. The SER may be reviewed to confirm the basis for NRC acceptance of the SAR." Section 3.5 provides as examples of changes that increase the probability of occurrence of a malfunction of equipment important to safety as "degrades below the design basis the performance of a safety system assumed to function in the accident analysis" and "increases challenges to safety systems assumed to function in the accident analysis such that safety system performance is degraded below the design basis without compensating effects." Finally, Section 4.2.3 "May the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the safety analysis report?" states that "after identifying the impact of the proposed activity on the important to safety (ITS) equipment, a determination is made if an increase in the probability of a malfunction of the ITS equipment has occurred." The following are selected examples of questions that can be used in making the determination:

> Will the proposed activity meet the original design specification for material and construction practices considering:
>
> - Are the seismic specifications met (such as use of proper supports, proper lugging at terminals, and isolation of lifted leads)?
>
> Will the proposed activity degrade SSC reliability by:
>
> - Imposing additional loads not analyzed in the original design?
>
> - Downgrading the support system performance necessary for reliable operation of ITS equipment?

Hence, the inspector concluded that the probability of occurrence of a malfunction was applicable to the licensee's seismic analysis for the removal of steam generator snubber seismic restraints (and indirectly for the subsequent steam generator replacement that proceeded without addressing this in a 10 CFR 50.59 evaluation because it was thought to be encompassed by the earlier modification). Specifically, the snubber seismic restraints that had been removed had prevented steam generator movement, but steam generator slider support friction does not prevent steam generator displacement during a seismic event. In addition, elimination of required conservatism inherent in the seismic licensing basis as noted above could make the RCS coupled system design with respect to the steam generator slider support more susceptible to failure from a seismic event than assumed in the NRC's acceptance of the seismic design for the reactor coolant system. In particular, use of the higher damping value resulted in less calculated component stress and displacement and hence could allow for a less effective seismic design than assumed in the licensing basis.

Further, Title 10 CFR 50.59(b)(1) (effective January 1, 1991) states, in part, that the licensee shall maintain records of changes in the facility to the extent that these changes in the facility constitute changes as described in the safety analysis report. These records must include a written safety evaluation which provides the bases for the determination that the change does not involve a USQ. The licensee's 10 CFR 50.59 evaluation, SE 91-0046, that supported removal of the steam generator supports, concluded that "the implementation of the items covered by this modification will not increase the probability of a malfunction of equipment important to plant safety because the reactor coolant system is maintained within its design requirements and in fact by this analysis safety-related equipment is shown to be not required and is removed, thereby, decreasing the probability of equipment malfunction."

However, the licensee's written safety evaluation failed to provide a suitable bases for the determination that the change did not involve a USQ in that it did not address how the basis for NRC's-approval of the seismic design of the reactor coolant system continued to be met. Specifically, the May 31, 1983, NRC safety evaluation approved the licensee's seismic analysis for reactor coolant system design because "there is sufficient conservatism and margin in the piping systems components and supports at Davis-Besse Unit 1 to ensure safe shutdown and continued shutdown heat removal in the event of an SSE having a ground acceleration of 0.20g." The licensee subsequently adopted a significantly higher damping value for the steam generator slider support while maintaining a 0.15g acceleration for the design without addressing how "sufficient conservatism and margin" otherwise continued to be met. And while the licensee's written safety evaluation did reference the NRC's approval of code relief to use ASME CC N-411 damping values, it did not address how "sufficient conservatism and margin" otherwise continued to be met (or provide rationale for exceeding the ASME CC N-411 allowed damping values for the steam generator slider support.).

The inspector did not make a determination whether the licensee's modifications constituted a USQ, only that the licensee had not adequately addressed that question in its 10 CFR 50.59 evaluation. The inspector would expect the licensee to make that determination in a subsequent 10 CFR 50.59 evaluation for the current design under the current rule that addresses pertinent aspects discussed above. In particular, the inspector could not eliminate the possibility of other calculation changes or the proper adoption of alternate methodologies the licensee might pursue that may not be contrary to the licensing basis and that would show that allowable stress limits continued to be met. In addition, the inspector did not make a determination that the licensee's damping value for the steam generator slider support was technically incorrect or not realistic, only that it was not conservative enough to meet the plant licensing basis that justified adoption of 0.15 g acceleration earthquake versus a stronger 0.2g acceleration earthquake for the plant design, and the licensee's 10 CFR 50.59 evaluation had not demonstrated how "sufficient conservatism and margin" continued to be met.

Consideration of Enforcement Discretion

The NRC Enforcement Manual, Revision 9, dated September 9, 2013, Part II-2, Section 2.1.3.E.5, provides for enforcement discretion for 10 CFR 50.59 situations that violate the "old" requirements, but that would not be violations had the evaluation been performed under the revised rule. Therefore, the inspector reviewed the change to RCS seismic licensing basis damping using the current rule and NRC endorsed guidance, NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1.

Title 10 CFR 50.59(a)(1) defines, in part, a change as a modification or addition to, or removal from, the facility that affects a design function or an evaluation that demonstrates that the intended function will be accomplished. Section 3.3 of NEI 96-07, in part, defines a change as a modification or addition to, or removal from, the facility or procedures that affects an evaluation that demonstrates that intended functions will be accomplished. In addition, Section 4.2.1.1, "Screening of Changes to the Facility as Described in the UFSAR," of NEI 96-07 provides the following guidance: "Changes are 'screened in' (i.e., require a 10 CFR 50.59 evaluation) if they adversely affect an SSC design function." Hence, the use of a damping value for the steam generator slider support in calculations for removal of the steam generator snubber seismic restraints that was substantially greater than the damping value established in the licensing basis would require a 10 CFR 50.59 evaluation be performed under the current rule.

Title 10 CFR 50.59(d)(1) requires the licensee to maintain records of changes in the facility and that these records must include a written safety evaluation which provides the basis for the determination that the change does not require a license amendment. Further, 10 CFR 50.59(c)(2)(ii) requires that the licensee shall obtain a license amendment if the change would 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 final safety analysis report (FSAR) (as updated). This parallels the old rule which requires, as noted above, that the licensee maintain records of changes in the facility that includes a written safety evaluation which provides the bases for the determination that the change does not involve a USQ, and that a USQ is involved if the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased. Hence, likelihood or probability of occurrence of a malfunction caused by the increase in damping value would also need to be specifically addressed in the 10 CFR 50.59 evaluation under the current rule with respect to the basis for approval in the May 31, 1983, NRC SER and corresponding statements in the USAR.

Further, 10 CFR 50.59(c)(2)(viii) requires the licensee obtain a license amendment if the change would result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety evaluation. The inspector noted that Example 1 of Section 4.3.8.3, "Does the Activity Result in a Departure from a Method of Evaluation Described in the UFSAR Used in Establishing the Design Basis or in the Safety Analysis?" of NEI 96-07, provided guidance for a change in USAR damping that would require prior NRC approval. Specifically, "using a higher damping value to represent the response of the piping to acceleration from the postulated earthquake in the analysis would result in lower calculated stresses because the increased damping reduces the loads. Since this analysis was used in establishing the seismic design bases for the piping, and since this is a change to an element of the method that is not conservative and is not essentially the same, this change would require prior NRC-approval under this criterion." Similarly, the licensee's use of higher damping values to represent the response of the RCS 1/2–Loop Model to the acceleration from the postulated earthquake was a change to an element of the method of analysis that was not conservative and was not essentially the same. Example 1 goes on to discuss NRC-approved alternate methods of seismic analysis that may allow higher damping, but the licensee's 10 CFR 50.59 evaluation, SE 91-0046, did not attempt to take advantage of that, as that option is specific to the current rule and associated NRC endorsed industry guidance.

Therefore, the licensee's 10 CFR 50.59 evaluation, SE 91-0046, would have been inadequate and not have met regulatory requirements under the new rule as well as the old rule. Hence, the inspector concluded that NRC Enforcement Manual Part II-2, Section 2.1.3.E.5 discretion criteria were not satisfied.

The licensee entered the issue of not performing a 10 CFR 50.59 evaluation for the change in USAR seismic licensing and design basis damping associated with seismic analysis of the RCS 1/2–Loop Model into the corrective action program as Condition Report (CR) 2016-10468, "NRC Inspection: 10 CFR 50.59 Safety Evaluation (SE 91-0046) Did Not Properly Document the Change to Seismic Licensing Basis," dated September 1, 2016.

Reactor Coolant System Operability

During inspection activities for the replacement steam generators, the inspector requested clarification regarding the seismic analysis methodology and regulatory basis for the RCS 1/2–Loop Model. Specifically, the inspector requested: a description of the modal weighted damping methodology; illustrations detailing how structures, piping, and components were modeled; and damping values used in the analysis. The inspector determined that licensee calculation 32-1177088-00 (snubber restraints removed at base of original steam generator) and calculation 32-9191681-002 (replacement steam generator) used seismic analysis methods described in ASME Section III, non-mandatory Appendix N, "Dynamic Analysis Methods," 1986 Edition through 1988 Addenda. Specifically, the calculations used Section N-1221.1.1, "Method of Modal Superposition," and modal damping based on N-1233.2, Equation 60. Although Section N-1230 recognized Coulomb damping (results from the sliding friction motion of a body on another surface), the inspector determined that Appendix N does not provide guidance to convert Coulomb damping (the damping force is proportional to the normal force) to equivalent viscous damping (the damping force is proportional to velocity) used in response spectrum analysis. The conversion of steam generator slider friction to an effective viscous damping was determined by calculation methods referenced in calculation 32-1177088-00 and calculation 32-9191681-002.

During inspection activities for the replacement steam generators, the licensee initiated CR-2014-07979 to address the RCS 1/2–Loop Model damping concerns. In this condition report, the licensee concluded that calculation 32-9191681-002 provided reasonable assurance the RCS piping and components will perform all their design basis functions during a seismic event because the calculation used ASME Section III Appendix N methods for seismic analysis and the calculated stresses were within code limits. The inspector used guidance in Inspection Manual Chapter (IMC) 0326, "Operability Determinations & Functionality Assessments for Conditions Adverse to Quality Or Safety," Section C.04, "Use of Alternate Analytical Methods in Operability

Determinations," to evaluate the licensee's operability conclusion and did not identify any concerns. In particular, IMC 0326, Section C.04, states: "The use of alternate methods is not subject to 10 CFR 50.59 unless the methods are used in the final corrective action" and the inspector concluded that the methodology used was a reasonable and technically sound approach. Further, after reviewing the calculations, the inspector agreed that there was reasonable assurance that the steam generator support structure would not fail during a 0.2g earthquake if realistic assumptions/inputs were applied.

Analysis: The inspector determined that the licensee's failure to provide in its 10 CFR 50.59 evaluation, SE 91-0046, a suitable basis for the determination that the use of damping higher than established in the seismic licensing basis for the reactor coolant system, specifically the steam generator slider support, was not a USQ was a performance deficiency. The issue of concern was determined to be more than minor because the performance deficiency impacted the Barrier Integrity cornerstone objective to provide reasonable assurance that physical design barriers (reactor coolant system) protect the public from radionuclide releases caused by accidents or events and design control attribute to maintain functionality of the RCS.

Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the Significance Determination Process (SDP). However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. The inspector determined the finding could be evaluated using the SDP in accordance with IMC 0609, "SDP," Attachment 0609.04, "Initial Characterization of Findings." In accordance with Table 2 for the Barrier Integrity Cornerstone, RCS boundary issues are considered under the Initiating Events Cornerstone. For Table 3 – SDP Appendix Router, the inspector answered 'No' to the questions, and therefore the finding was evaluated using the SDP in accordance with IMC 0609, "The SDP for Findings at Power," Appendix A, Exhibit 1, "Initiating Events Screening Questions." The inspector answered "No" to all the questions in Exhibit 1. In particular, because the reactor coolant system remained operable (capable of performing its safety function during a seismic event), the finding was determined to have very low safety significance (Green) corresponding to a Severity Level IV violation per Example 6.1.d.2 of the NRC Enforcement Policy.

The inspector determined the finding did not have a cross-cutting aspect because it is not representative of current licensee performance. Specifically, the failure to perform an adequate 10 CFR 50.59 evaluation for the change in seismic licensing basis damping for the steam generator slider support used in the RCS 1/2-Loop coupled system model occurred in 1991 to support snubber removal. Subsequently, the licensee did not evaluate this change while performing its 10 CFR 50.59 screening, 12-03404, for the 2014 steam generator replacement, because the supporting calculation used damping that was actually more conservative than that used in this earlier re-analysis.

Enforcement

Title 10 CFR 50.59(b)(1) (version effective January 1, 1991) states, in part, that the licensee shall maintain records of changes in the facility to the extent that these changes in the facility constitute changes as described in the safety analysis report. These records must include a written safety evaluation which provides the bases for the determination that the change does not involve a USQ. In addition, 10 CFR 50.59(a)(2) specifies, in part, that a proposed change shall be deemed to involve a USQ if one or more specific criteria is met, one of which is if the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased. Davis-Besse USAR Section 3.7.2.14, "Coupled System Damping," stated: "Although various components within a model possess different values of critical damping (example: shield wall – 2 percent, reactor, steam generators, and pressurizer – 1 percent, and piping 1/2 percent), the lowest damping value was used for all components in the model by the NSSS supplier."

Contrary to the above, on approximately August 6, 1991, the licensee made changes to the facility as described in the safety analysis report: in particular, the licensee permanently disconnected snubbers at the base of the steam generators (USAR Change Notice 91-057 revised Figure 3.9-1, "DBNPS, Once Through Steam Generator and Wall Dynamic Model") to conform to plant Modification 90-0079 and changed the original seismic licensing basis damping defined in Section 3.7.2.14 for coupled system damping (a conservative 1/2 percent damping value was used to develop seismic spectra based on a 0.15g SSE ground acceleration). In doing so, the licensee failed to maintain records that included a written safety evaluation which provided the bases for determining that the change to seismic licensing basis damping in calculations to support removal of snubbers under modification 90-0079 did not involve an unreviewed safety question. Specifically, licensee safety evaluation SE 91-0046 did not provide a suitable basis for concluding that there was no increase in the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report, in that it did not address how the basis for the NRC's-approval of the seismic design of the reactor coolant system continued to be met with respect to the steam generator slider support (Lubrite plate) damping. In particular, a May 31, 1983, NRC Safety Evaluation Report approved the licensee's use of a 0.15g acceleration earthquake in its seismic analysis for reactor coolant system design, in part, because "there is sufficient conservatism and margin in the piping systems components and supports at Davis-Besse Unit 1 to ensure safe shutdown and continued shutdown heat removal in the event of an SSE having a ground acceleration of 0.20g." The licensee subsequently adopted a significantly higher damping value for the steam generator slider support while maintaining a 0.15g acceleration for the design without addressing how "sufficient conservatism and margin" otherwise continued to be met. Because this violation was of very-low safety significance and was entered into the licensee's corrective action program as CR-2016-10468, this Severity Level IV violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy (**NCV 05000346/2016010-01, 10 CFR 50.59 Evaluation Failed to Consider Change to Seismic Licensing Basis**).

Based on the above discussion, URI 5000346/2013010-01 is closed.

4OA6 Management Meetings

.1 Exit Meeting Summary

On September 7, 2016, the inspector presented the inspection results to the Site Vice President, Mr. B. Boles, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector confirmed with the licensee the scope of material reviewed that was considered to be proprietary. Proprietary information reviewed by the inspector was controlled in accordance with appropriate NRC policies regarding sensitive unclassified information, and has been denoted as “proprietary” in the Attachment to this report.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Boles, Site Vice President
K. Byrd, Director, Site Engineering
D. Blakely, Supervisor, Reactor Engineering
J. Hook, Design Engineering
P. McCloskey, Manager, Site Regulatory Compliance
G. Michael, Manager, Design Engineering
T. Ridlon, Design Engineering
J. Sturdavant, Regulatory Compliance
G. Wolf, Supervisor, Regulatory Compliance

U.S. Nuclear Regulatory Commission

D. Kimble, Senior Resident Inspector
T. Briley, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000346/2016010-01	NCV	10 CFR 50.59 Evaluation Failed to Consider Change to Seismic Licensing Basis (Section 4OA5.1)

Closed

05000346/2016010-01	NCV	10 CFR 50.59 Evaluation Failed to Consider Change to Seismic Licensing Basis (Section 4OA5.1)
05000346/2013010-01	URI	10 CFR 50.59 Safety Evaluation Not Performed for Calculated Component Damping (Section 4OA5.1)

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather, that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

4OA5 Other Activities

- 10 CFR 50.59 Safety Evaluation; FCR 84-002, Supplement 1: ASME CC N-411; Dated February 28, 1985
- Calculation 32-1177088-06; TE 1/2-Loop Model Reanalysis; Revision 6 [PROPRIETARY]
- Calculation 32-9191681-002; DB-1 RCS 1/2-Loop Model and Loading Analysis; Revision 2 [PROPRIETARY]
- CR 2014-07979; TE 1/2-Loop Model Reanalysis – AREVA Calculation 32-1177088-00; Originated April 30, 2014
- DBNPS Final Safety Analysis Report, Volume 2; Table 3-7, Percentage of Critical Damping Factors; Revision 10
- DBNPS Preliminary Safety Analysis Report, Volume 2; Table 5A-1, Percentage of Critical Damping Factors; Revision 0
- DBNPS Updated Safety Analysis Report; Revision 30
- NG-EN-00304; Toledo Edison Nuclear Group Procedure: Safety Review and Evaluation; Revision 1 (effective January 18, 1990)
- NRC Letter to Toledo Edison Company; License Amendment No. 30: Deletes Satisfied License Condition 2.C.(3)(r); Dated August 27, 1980 (ADAMS Accession No. ML021200322)
- NRC Letter to Toledo Edison Company; Subject: License Condition Guidelines for Seismic Reanalysis of Piping Systems and Components; Dated January 30, 1979
- NRC Letter to Toledo Edison Company; Subject: Seismic Reanalysis of Piping Systems and Components; Dated May 31, 1983
- NRC Letter to Toledo Edison Company; Subject: Use of ASME CC N-411; Dated April 15, 1985
- NRC Letter to Toledo Edison Company; Subject: Use of ASME CC N-411; Dated August 26, 1985
- NSAC/125; Guidelines for 10 CFR50.59 Safety Evaluations; Dated June 1989
- NUREG-0136; Safety Evaluation Report Related to Operation of Davis-Besse Nuclear Power Station Unit 1; Dated December 1976
- NUREG-0136, Supplement 1; Safety Evaluation Report Related to Operation of Davis-Besse Nuclear Power Station Unit 1; Dated April 1977
- SE 99-0099; 10 CFR 50.59 Safety Evaluation for UCN 89-026; Dated May 5, 1989
- SE 91-0046; 10 CFR 50.59 Safety Evaluation for Modification 90-0079: Permanent Removal of Seismic Restraints (Snubbers); Revision 1; Dated August 6, 1991
- Stress Analysis for Toledo Edison Company (NSS-14) Primary Code Piping; Dated April 19, 1972 [PROPRIETARY]
- Toledo Edison Letter to NRC, Serial No. 525; Seismic Reanalysis and Evaluation Submittal; Dated July 9, 1979
- Toledo Edison Letter to NRC, Serial No. 614; Supplement 1 to Seismic Reanalysis and Evaluation Submittal dated July 9, 1979 (Serial No. 525); Dated May 16, 1980
- Toledo Edison Letter to NRC, Serial No. 630; Supplement 2 to Seismic Reanalysis and Evaluation Submittal dated July 9, 1979 (Serial No. 525); Dated July 22, 1980

- Toledo Edison Letter to NRC, Serial No. 695; Revision 1 to Supplement 2 to Seismic Reanalysis and Evaluation Submittal dated July 9, 1979 (Serial No. 525); Dated March 13, 1981
- Toledo Edison Letter to NRC, Serial No. 1129; Request for NRC Approval to Use ASME CC N-411 Alternate Damping Values for Seismic Analysis of Piping; Dated March 14, 1985
- Toledo Edison Letter to NRC, Serial No. 1163; Request for NRC Approval to Use ASME CC N-411 Alternate Damping Values for Seismic Analysis of Piping; Dated July 2, 1985
- UCN 89-026; USAR Change Notice: Reconcile Damping Differences between Davis-Besse USAR and Design Criteria Manual; Dated April 24, 1989
- UCN 91-057; USAR Change Notice: Modification 90-0079; Dated August 1992

Corrective Action Documents Generated Due to the Inspection

- CR 2016-10468; NRC Inspection: 10 CFR 50.59 Safety Evaluation (SE 91-0046) Did not Properly Document the Change to Seismic Licensing Basis; Originated September 1, 2016

LIST OF ACRONYMS USED

ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
ASME	American Society of Mechanical Engineers
CC	Code Case
CFR	*Code of Federal Regulations*
CR	Condition Report
DBNPS	Davis-Besse Nuclear Power Station
FSAR	Final Safety Analysis Report
IMC	Inspection Manual Chapter
ITS	Important to Safety
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
OBE	Operating Basis Earthquake
PARS	Publicly Available Records System
RCS	Reactor Coolant System
RG	Regulatory Guide
SAR	Safety Analysis Report
SDP	Significance Determination Process
SE	Safety Evaluation
SER	Safety Evaluation Report
SSC	Structure, System, or Component
SSE	Safe Shutdown Earthquake
URI	Unresolved Item
UFSAR	Updated Final Safety Analysis Report
USAR	Updated Safety Analysis Report
USQ	Unreviewed Safety Question

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Sincerely,

/RA/

David E. Hills, Chief
Engineering Branch 1
Division of Reactor Safety

Docket No. 50–346
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