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10 CFR 50.55a

RS-22-046

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Subject: Relief Requests Associated with the Sixth Inservice Inspection Interval

In accordance with 10 CFR 50.55a, "Codes and standards," Paragraph (z)(2), Constellation Energy Generation, LLC (CEG) requests NRC approval of the attached relief requests associated with the Sixth Inservice Inspection (ISI) Interval for Dresden Nuclear Power Station (DNPS), Units 2 and 3. The Sixth Interval of the DNPS, Units 2 and 3, ISI Program will comply with the 2017 Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code.

The Sixth ISI Interval at DNPS will begin on January 20, 2023, and is currently scheduled to end January 19, 2033. Accordingly, CEG requests approval of these requests by January 20, 2023, to support the Sixth ISI Interval.

There are no regulatory commitments contained within this letter. Should you have any questions concerning this letter, please contact Mr. Mitch Mathews at (630) 657-2819.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Sr. Manager – Licensing
Constellation Energy Generation, LLC

Attachment: Relief Requests Associated with the Sixth Inservice Inspection Interval

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – Dresden Nuclear Power Station

ATTACHMENT
Relief Requests Associated with the Sixth Inservice Inspection Interval

A listing of the relief requests associated with the Sixth Inservice Inspection interval for Dresden Nuclear Power Station, Units 2 and 3 is provided below.

Designator	Description	Comments
I6R-01	Relief for Inspection of Standby Liquid Control Nozzle Inner Radius	Approved for Fifth ISI Interval
I6R-02	Alternative for Testing Frequency for Isolation Condenser Shell Side and Associated Piping	Approved for Fifth ISI Interval

**10 CFR 50.55a Relief Request I6R-01,
Relief for Inspection of Standby Liquid Control Nozzle Inner Radius in Accordance with 10 CFR 50.55a(z)(2) -- Hardship or Unusual Difficulty Without Compensating Increase in the Level of Quality and Safety -- Revision 0**

1. ASME Code Component(s) Affected

Code Class:	1
Reference:	IWB-2500, Table IWB-2500-1
Examination Category:	B-D
Item Number:	B3.100
Description:	Inspection of Standby Liquid Control Nozzle Inner Radius
Component Number:	Unit 2: N12-1 Unit 3: N12-1
Drawing Number:	Unit 2: ISI-128, Sheet 3 Unit 3: ISI-125, Sheet 3

2. Applicable Code Edition

The Sixth Interval of the Dresden Nuclear Power Station (DNPS), Units 2 and 3 Inservice Inspection (ISI) Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI, 2017 Edition.

3. Applicable Code Requirement

IWB-2500 states that components shall be examined and tested as specified in Table IWB-2500-1.

Table IWB-2500-1 requires a volumetric examination to be performed on the inner radius section of all reactor pressure vessel nozzles each inspection interval.

4. Reason for Request

In accordance with 10 CFR 50.55a(z)(2), relief is requested on the basis that conformance with the specified Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The Standby Liquid Control (SBLC) nozzle, as shown in Figure I6R-01.1, is designed with an integral socket to which the boron injection piping is welded. The SBLC nozzle is located in the upper segment of the lower head of the reactor pressure vessel in an area that is inaccessible for examination from the inside of the reactor pressure vessel. Therefore, the examination must be performed from the outside surface of the lower reactor pressure vessel head.

An examination from the outside surface is conducted from either the nozzle-to-head outer blend radius or the head (plate) surface near the nozzle. Both examination areas provide a unique approach and are discussed separately.

Because of the small diameter of the SBLC nozzle (i.e., nominal two (2) inches) and the thickness of the lower reactor vessel head (i.e., nominal 6.25 inches), the ratio of the nozzle diameter to the head thickness prevents a meaningful examination from the nozzle-to-head outer blend radius. The inside surface inspection angle, utilizing specialized/contoured wedges or shoes, needs to be a mid-range tangential angle (i.e., 30 degrees - 60 degrees) to adequately detect flaws at the inside radius area of interest while scanning from the outer blend. Given the small diameter nozzle and utilizing specialized / contoured wedges or shoes, the angle will be in the low-range of 0 degrees – 15 degrees (i.e., essentially a straight beam) when focused to tangentially coincide with the inner radius area of interest. This low angle will not provide sufficient reflectivity for detection of inservice induced defects.

An examination from the head (plate) surface near the nozzle is also difficult, since the forged nozzle has a stainless steel cladding welded to the inside surface with increased weldment applied at the inner radius area for construction of an engineered socket configuration. After final machining, this engineered socket receives an austenitic internal pipe, which is integrally welded to form a complex cladding/socket configuration. The geometry and change in grain structures at the dissimilar material interface prohibit a timely (i.e., short duration) and meaningful ultrasonic examination of the inside radius section of the nozzle.

Even if a meaningful examination could be completed, the change in grain structures result in mode conversion and angle changes, consequently, significantly longer examination duration is required to resolve a variety of reflective signals. Examination of this unique design is estimated to require a minimum of two hours with a two-person team (i.e., four man-hours) while a typical nozzle inner radius examination will require up to one man-hour with a two-person team.

A review of current ultrasonic techniques was conducted including discussions with the Electric Power Research Institute (EPRI). The long ultrasonic metal path and potential for multiple geometric and dissimilar material reflectors inherent in the nozzle design prevent a meaningful examination from being performed on the inner radius of the SBLC nozzle.

Ultrasonic examination of this configuration will require the use of multiple techniques, modes, angles, and an extensive amount of time on the component for best effort signal discrimination between geometrical and flaw type signals. This unique design is not typical for designs referenced in ASME Section XI, but is more closely related to a dissimilar metal component. At present, the industry has a qualified technique for dissimilar metal piping welds, but has not addressed this unique design.

The dose exposure to the technicians performing the examination is a relevant concern. While actions would be taken to provide protection against the inherent radiation field, the large dose rates in the lower head region near the SBLC nozzle present a unique challenge. Dose rates in this region can be up to 40 millirem (mrem) per hour. As such, radiation exposure for this examination could easily exceed 160 mrem, based on the general area dose and examination duration.

The proposed relief request will not compromise the level of quality and safety. The inner radius socket attaches to piping that delivers the boron solution far away from the nozzle inner radius. Therefore, the SBLC nozzle inner radius section is not subjected to turbulent mixing conditions that are a concern with other penetrations. In addition, a VT-2 visual examination of the SBLC nozzle is part of the Class 1 system leakage test scheduled during each refueling outage. This VT-2 visual examination is an ASME Section XI-required examination that verifies structural integrity and leak tightness for the entire Class 1 boundary. During reactor operation, the SBLC nozzle is inaccessible for direct visual observation based on its location in the Drywell. This does not preclude the nozzle from being monitored. Potential leakage from the nozzle would be collected in the Drywell sumps, prompting action from the station. Technical Specifications (TS) requirements ensure Reactor Coolant System (RCS) leakage is regularly monitored (i.e., every 12 hours), and specify prudent actions, up to and including reactor shutdown, should predetermined limits be exceeded. These actions ensure that the overall level of plant quality and safety will not be compromised. Historically, no leakage from this nozzle has ever been observed at DNPS from this location.

Relief is requested in accordance with 10 CFR 50.55a(z)(2), since compliance with the applicable Code requirements would require an ultrasonic examination to be performed on the outside diameter of the reactor pressure vessel. Geometric and material reflectors would prevent a meaningful examination, resulting in inaccurate data. Based on this, the specified Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety in accordance with 10 CFR 50.55a(z)(2).

5. <u>Proposed Alternate and Basis for Use</u>

A VT-2 visual examination of the SBLC nozzles at DNPS, Units 2 and 3 will be part of the scheduled Class 1 system leakage test performed each refueling outage. In addition, the TS Surveillance Requirement (SR) for RCS leakage will be satisfied during plant operation (i.e., TS SR 3.4.4.1). These actions ensure that the overall level of plant quality and safety will not be compromised.

6. <u>Duration of Proposed Alternative</u>

Relief is requested for the Sixth ISI Interval for DNPS, Units 2 and 3

7. <u>Precedents</u>

- DNPS, Units 2 and 3 Fifth ISI Interval Relief Request I5R-01, Revision 1 was authorized by NRC safety evaluation (SE) dated September 30, 2013 (NRC Accession No. ML13260A585). The Sixth ISI Interval Relief Request utilizes a similar approach that was previously authorized

- Quad Cities Nuclear Power Station, Units 1 and 2, Fifth ISI Interval Relief Request I5R-01, Revision 1 was authorized by NRC SE dated September 30, 2013 (NRC Accession No. ML13267A097)

8. References

None

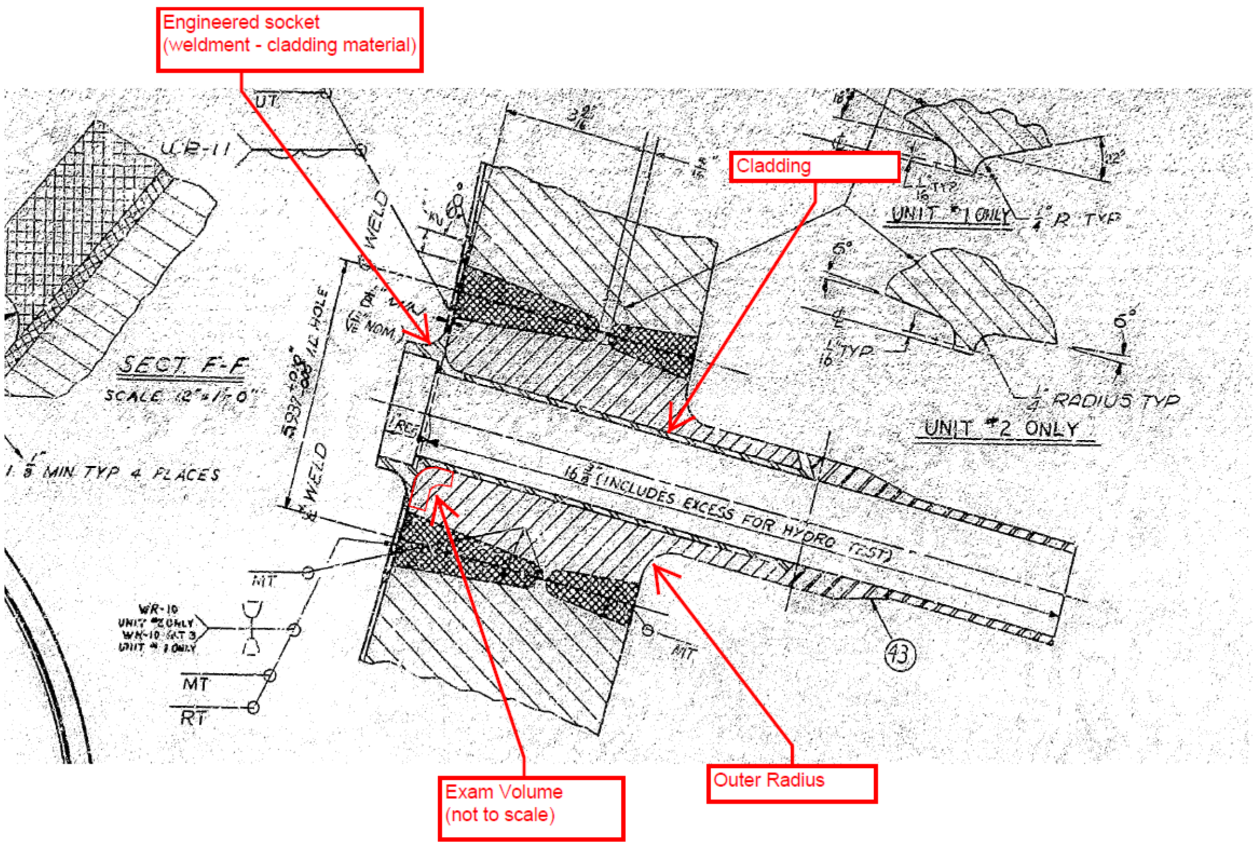

Figure I6R-01.1: TWO INCH STANDBY LIQUID CONTROL NOZZLE

10 CFR 50.55a Relief Request I6R-02,
Alternative for Testing Frequency for Isolation Condenser Shell Side and Associated Piping in Accordance with 10 CFR 50.55a(z)(2) -- Hardship or Unusual Difficulty Without Compensating Increase in the Level of Quality and Safety -- Revision 0

1. ASME Code Component(s) Affected

Code Class:	3
Reference:	IWD-2500, Table IWD-2500-1
Examination Category:	D-B
Item Number:	D2.10
Description:	Testing Frequency for Isolation Condenser Shell Side and Associated Piping

Component Number:

Unit No.	Drawing	Test Block No.
2	M-28, M-39	2IC01, 2IC02
3	M-359, M-369	3IC01, 3IC02

2. Applicable Code Edition

The Sixth Interval of the Dresden Nuclear Power Station (DNPS), Units 2 and 3 Inservice Inspection (ISI) Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI, 2017 Edition.

3. Applicable Code Requirements

Table IWD-2500-1, "Examination Categories," Examination Category D-B, Item Number D2.10, requires all Class 3 pressure retaining components be subject to a system leakage test with a VT-2 visual examination in accordance with IWD-5220, "System Leakage Test." This system leakage test is to be conducted once each inspection period.

IWD-5221(b), "Pressure," states for Class 3 [Table IWD-2500-1 (D-B)] components in standby systems (or portions of standby systems) that are not operated routinely except for testing, the leakage test shall be conducted at the system pressure developed during a test conducted to verify system operability (e.g., to demonstrate system safety function or satisfy technical specification surveillance requirements). If portions of a system are associated with more than one safety function, the visual examination need only be performed during the test conducted at the higher of the test pressures for the respective system safety function.

4. Reason for Request

In accordance with 10 CFR 50.55a(z)(2), relief is requested on the basis that conformance with the specified Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The Isolation Condenser (IC) is not normally in service; it is normally in a standby alignment with its shell side vented to the atmosphere through a non-isolable vent line. For inservice inspection purposes, each unit's IC is divided into two test blocks (i.e., Test Blocks 2IC01 and 2IC02 for the DNPS, Unit 2, IC upper and lower portions, respectively, and Test Blocks 3IC01 and 3IC02 for the DNPS, Unit 3, IC upper and lower portions, respectively). Figures I6R-02.1 and I6R-02.2 below provide a representation of the upper and lower system leakage test blocks for Unit 2.

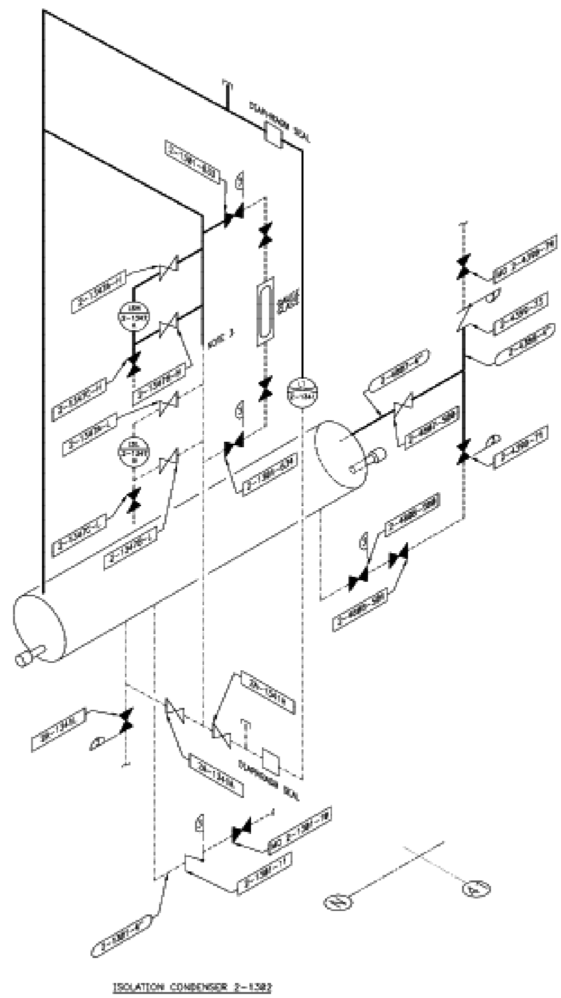

Figure I6R-02.1: Unit 2 Isolation Condenser Test Block No. 2IC01

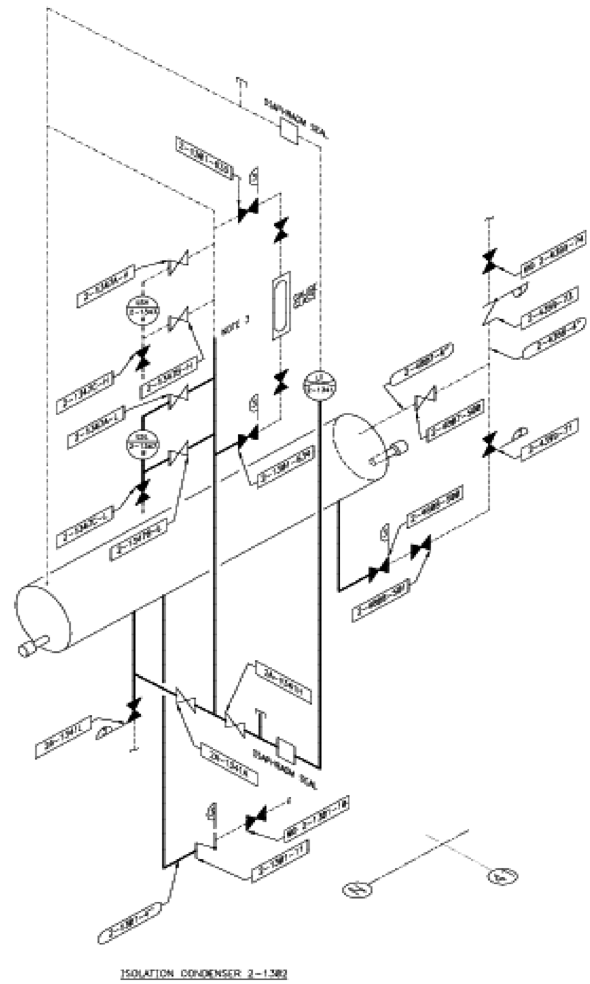

Figure I6R-02.2: Unit 2 Isolation Condenser Test Block No. 2IC02

The system is normally aligned with the IC shell side water level greater than or equal to six (6) feet in accordance with DNPS, Units 2 and 3 Technical Specifications (TS).

Constellation Energy Generation (CEG), LLC has previously submitted and received approval to conduct the system leakage test of the IC shell and associated piping during performance of TS Surveillance Requirement (SR) 3.5.3.4 instead of at the ASME Section XI-required frequency of once per inspection period. TS SR 3.5.3.4 is performed in accordance with the Surveillance Frequency Control Program (SFCP) at a 120-month frequency.

The system pressure developed during the performance of this TS SR meets the ASME Section XI requirements of IWD-5221(b); however, the TS Frequency of 120-months (i.e., once per ISI Interval) does not meet the Table IWD-2500-1 examination frequency requirement of once per inspection period.

Performance of additional IC heat removal capability tests solely for the purposes of performing a system leakage test requires a minimum of a 25% reduction in reactor power to perform the examination. This introduces an unnecessary transient on the affected DNPS reactor and a challenge to station operators. During actuation of the IC, one valve is opened to allow condensate in the IC tube bundle to return back to the reactor vessel. As a result, a volume of relatively cold water is returned to the reactor resulting in an increase in reactor power. In addition, Unit 3 utilizes Reactor Water Cleanup (RWCU) system return piping for the IC condensate return path which requires the RWCU system to be removed from service prior to performing TS SR 3.5.3.4 and returned to service following the performance of the SR. This is necessary to mitigate the thermal transient on the RWCU piping. Manipulation of the RWCU system does present a small transient to reactor operation and has even resulted in an automatic reactor scram on a low reactor water level signal during the restoration of RWCU system operation on one occasion at DNPS.

During the performance of IC heat capacity testing, dose rates increase up to 100 millirem per hour (mR/hr) (i.e., normal floor dose rates are <5 mR/hr) on the IC floor during the test. This challenges radiological safety of station personnel during the performance of the VT-2 visual examination while the IC is in service. The total personnel dose received during the performance of the IC system leakage test is typically about 125 millirem.

During the IC heat removal test, shell side water is used to condense reactor steam in the tube bundle. The shell side water volume boils and is exhausted through the IC vent pipe that extends through the Reactor Building wall and discharges to the local atmosphere. For the safety of plant personnel, access to the vicinity of the IC vent must be controlled during the performance of the IC heat capacity tests.

As previously stated, the IC shell cannot be isolated and pressurized to meet IWD-5221(b) examination pressure requirements when in a standby alignment. Moreover, it would be an abnormal activity to fill the IC to the top simply to achieve a slight increase in static head for the additional system leakage test. As an additional complication, water added to the IC shell to raise level above the normal standby conditions would subsequently have to be drained and processed as radwaste.

In summary, imposing a transient on the reactor plant to accommodate the performance of the IC heat removal capability verification at a greater frequency than required by DNPS, Units 2 and 3 TS SR 3.5.3.4, presents hardship without a compensating increase in the level of quality and safety.

5. Proposed Alternate and Basis for Use

As an alternative, CEG proposes the performance of a system leakage test using a VT-2 visual examination of all DNPS, Units 2 and 3 IC Test Blocks (i.e., Test Blocks 2(3)IC01 and 2(3)IC02) during the performance of TS SR 3.5.3.4, every 120 months (i.e., 10 years). The provisions of TS SR 3.0.2 are applicable to the TS SR 3.5.3.4 Frequency; therefore, the Frequency extension allowances of TS SR 3.0.2, may also be applied to future performances of TS SR 3.5.3.4, which could also impact the scheduling of future IC system leakage tests.

During inspection periods where TS SR 3.5.3.4 is not performed on a unit's IC, VT-2 visual examinations will be performed for Test Blocks 2(3)IC02 (i.e., the lower portion of the IC and associated piping). These leakage tests will be performed with IC shell side water level at the normal standby level versus at the normal pressure when the system is in service performing its operating function or at the system pressure developed during a test conducted to verify system operability as discussed in IWD-5221(b).

CEG proposes that the leakage test of IC Test Blocks 2(3)IC01 (i.e., the upper portion of the IC and associated piping) will only be performed once every inspection interval during the performance of TS SR 3.5.3.4 versus once per inspection period as discussed in Table IWD-2500-1. In the other two inspection periods, a VT-2 visual examination will be performed for test blocks 2(3)IC02 using the static head pressure associated with the IC level maintained during normal standby conditions.

6. Duration of Proposed Alternative

Relief is requested for the Sixth ISI Interval for DNPS, Units 2 and 3.

7. Precedent

DNPS, Units 2 and 3 Fifth ISI Interval Relief Request I5R-04, Revision 2 was authorized in an NRC safety evaluation (SE) dated January 13, 2020 (NRC Accession No ML20008D276). The Sixth ISI Interval Relief Request utilizes a similar approach that was previously authorized.