Exelon Generation

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September 28, 2012

10 CFR 50.55a

RS-12-151

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

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

Subject: Dresden Nuclear Power Station, Units 2 and 3, Fifth Interval Inservice Inspection Program Plan and Relief Requests

In accordance with 10 CFR 50.55a, "Codes and standards," paragraph (g), "Inservice inspection requirements," Exelon Generation Company, LLC (EGC) is required to update the Dresden Nuclear Power Station (DNPS), Units 2 and 3, Inservice Inspection (ISI) Program as described in the American Society of Mechanical Engineers (ASME) Section XI (the Code) once every 120-month inspection interval. Specifically, the ISI program is required to comply with the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a 12 months prior to the start of the interval in accordance with 10 CFR 50.55a(g)(4)(ii). The next interval for DNPS Units 2 and 3 is currently scheduled to commence on January 20, 2013, and scheduled to be completed by January 19, 2023. Accordingly, the 2007 Edition through the 2008 Addenda of ASME Section XI is the Code that DNPS will implement for the Fifth Interval Ten-Year ISI Program.

EGC is submitting the Fifth Interval Ten-Year ISI Program Plan (see Attachment A) in accordance with IWA-1400, "Owners Responsibility," paragraph (c) of the Code. Please note that EGC is requesting NRC approval of only the relief requests contained in Section 8, "Relief Requests from ASME Section XI." The remaining sections of the ISI Program Plan are provided for information only. The DNPS ISI Program Plan contains the following sections:

Section 1: Introduction and Background
Section 2: Basis for Inservice Inspection Program
Section 3: Component ISI Plan
Section 4: Support ISI Plan
Section 5: System Pressure Testing ISI Plan
Section 6: Containment ISI Plan
Section 7: Component Summary Tables
Section 8: Relief Requests from ASME Section XI
Section 9: References

Where alternatives to Code requirements are being proposed (i.e., in accordance with 10 CFR 50.55a(a)(3)(i) and (ii)), or the implementation of certain Code requirements has been determined to be impractical (i.e., in accordance with 10 CFR 50.55a(g)(5)(iii)), specific relief requests have been included in Section 8 of the Fifth Interval ISI Program Plan. Note that Relief Requests I5R-01, 02, 03, 04, 05 and 07 are similar to those previously approved for the Fourth Ten-Year Inspection Interval ISI Program. Relief Requests I5R-06 and I5R-09 have been previously authorized for the Fifth Ten-Year Inspection Interval and are included in this ISI Program Plan for completeness only. No further action by the NRC is requested for these two relief requests. Relief Requests I5R-10 and I5R-11 are new relief requests for the upcoming 10-year interval. Relief Request I5R-08 is reserved for future use.

The subject relief requests deal with the implementation of certain ISI program requirements which affect components to be examined during the Unit 2 November 2013 refueling outage and the Unit 3 October 2014 refueling outage. Approval of these relief requests in the customary one-year period (i.e., by September 28, 2013) will be sufficient to support these outages.

Should you have any questions concerning this letter, please contact Joseph A. Bauer at 630-657-2804.

Respectfully,

David M. Gullott
Manager – Licensing
Exelon Generation Company, LLC

Attachment A: Dresden Nuclear Power Station, Units 2 and 3, Inservice Inspection Program Plan, Fifth 10-Year Inspection Interval

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

ATTACHMENT A

Dresden Nuclear Power Station, Units 2 and 3
Inservice Inspection Program Plan
Fifth Ten-Year Interval

Exelon Generation Company
Dresden Nuclear Power Station
Units 2 & 3

ISI Program Plan

Fifth Ten-Year Inspection Interval

Report: DRE-483097-RP03
Rev: 0

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REVISION APPROVAL SHEET

TITLE: ISI Program Plan
Fifth Ten-Year Inspection Interval
Dresden Nuclear Power Station, Units 2 & 3

REVISION: 0

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EXELON ACCEPTANCE

APPROVED: John N. Kish / 9/19/12
John N. Kish
Inservice Inspection Owner/Responsible Individual

REVISION APPROVAL SHEET

TITLE: ISI Program Plan
Fifth Ten-Year Inspection Interval
Dresden Nuclear Power Station, Units 2 & 3

REVISION: 0

PROGRAM ACCEPTANCE

REVIEWED: John N. Kish / 9/19/12
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Each time this document is revised, the Revision Control Sheet should be completed to provide a detailed record of the revision history. The major changes should be outlined within the table. Editorial and formatting revisions are not required to be logged. The signatures above apply only to the changes made in the revision noted. If historical signatures are required, Dresden Nuclear Power Station archives will need to be retrieved.

REVISION CONTROL SHEET

Major changes should be outlined within the table below. Minor editorial and formatting revisions are not required to be logged.

Revision	Date	Revision Summary
0	9/19/12	Initial Issue. (Developed by Automated Engineering Services Corporation as part of the Dresden Nuclear Power Station Fifth Interval ISI Program Update.) Prepared: S. Coleman/M. King Reviewed: K. Johnson Approved: D. Lamond

Note: This ISI Program Plan (Sections 1 - 9 inclusive) is controlled by the Dresden Nuclear Power Station Engineering Programs Group.

REVISION SUMMARY

Section	Effective Pages	Revision	Date
Preface	i to vii	0	9/19/12
1.0	1-1 to 1-17	0	9/19/12
2.0	2-1 to 2-29	0	9/19/12
3.0	3-1 to 3-2	0	9/19/12
4.0	4-1 to 4-2	0	9/19/12
5.0	5-1	0	9/19/12
6.0	6-1 to 6-3	0	9/19/12
7.0	7-1 to 7-38	0	9/19/12
8.0	8-1 to 8-3	0	9/19/12
9.0	9-1 to 9-4	0	9/19/12

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1.0 INTRODUCTION AND BACKGROUND

1.1 Introduction

This Inservice Inspection (ISI) Program Plan details the requirements for the examination and testing of ISI Class 1, 2, 3, and MC pressure retaining components, supports, and containment structures at Dresden Nuclear Power Station (DNPS), Units 2, 3, and Common (2/3). Unit Common components are included in the Unit 3 sections, reports, and tables. This ISI Program Plan also includes Containment Inservice Inspection (CISI), Risk-Informed Inservice Inspections (RISI), Augmented Inspections (AUG), and System Pressure Testing (SPT) requirements imposed on or committed to by DNPS. This ISI Program Plan is controlled and revised in accordance with the requirements of procedure ER-AA-330, “Conduct of Inservice Inspection Activities,” which implements the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI ISI Program. At DNPS, the Inservice Testing (IST) Program is maintained and implemented separately from the ISI Program. The IST Basis Document and IST Program Plan contain all applicable inservice testing requirements. Procedure ER-AA-321, “Administrative Requirements for Inservice Testing,” implements the IST Program. The Snubber Program is also maintained and implemented separately from the ISI Program at DNPS. The Snubber Program documents contain all of the applicable snubber visual examination, functional testing, and service life monitoring requirements. The ISI Program Plan is credited as the existing program for DNPS License Renewal Aging Management Programs.

The Fifth ISI Interval is effective from January 20, 2013, through January 19, 2023, for DNPS Units 2 and 3. The Second CISI Interval is effective from September 9, 2008, through September 8, 2018, for DNPS Units 2 and 3. These effective interval dates are based on the fact that DNPS has been approved to extend plant operation under the license renewal application. The ASME Section XI Code of Record for the Fifth ISI Interval is the 2007 Edition through the 2008 Addenda, and the ASME Section XI Code of Record for the Second CISI Interval is the 2001 Edition through the 2003 Addenda.

Paragraph IWA-2430(c)(1) of ASME Section XI allows an inspection interval to be extended or decreased by as much as one year, and Paragraph IWA-2430(d) allows an inspection interval to be extended when a unit is out of service continuously for six months or more. The extension may be taken for a period of time not to exceed the duration of the outage. See Tables 1.1-1 and 1.1-2 for intervals, periods, and extensions that apply to DNPS’s Fifth ISI Interval and Second CISI Interval.

The Fifth ISI Interval and Second CISI Interval are divided into three successive inspection periods as determined by calendar years of plant service within the inspection interval. Tables 1.1-1 and 1.1-2 identify the period start and end dates for the Fifth ISI Interval and Second CISI Interval as defined by the Inspection Program. In accordance with Paragraph IWA-2430(c)(3), the inspection periods specified in these Tables may be decreased or extended by as much as 1 year to enable inspections to coincide with DNPS's refueling outages.

TABLE 1.1-1
UNITS 2 & 3 FIFTH ISI INTERVAL/PERIOD/OUTAGE MATRIX
(FOR ISI CLASS 1, 2, AND 3 COMPONENT EXAMINATIONS)

Unit 2		**Period**	**Interval**	**Period**	**Unit 3**	
Outage Number	**Projected Outage Start Date or Outage Duration**	**Start Date to End Date**	**Start Date to End Date**	**Start Date to End Date**	**Projected Outage Start Date or Outage Duration**	**Outage Number**
D2R23	Scheduled 11/13	1st 1/20/13 to 1/19/16	5th (Unit 2) 1/20/13 to1/19/23 5th (Unit 3) 1/20/13[1] to 1/19/23	1st 1/20/13 to 1/19/16	Scheduled 11/14	D3R23
D2R24	Scheduled 11/15			2nd 1/20/16 to 9/30/19[2]	Scheduled 11/16	D3R24
D2R25	Scheduled 11/17	2nd 1/20/16 to 9/30/19[2]			Scheduled 11/18	D3R25
D2R26	Scheduled 11/19	3rd 10/1/19 to 1/19/23		3rd 10/1/19 to 1/19/23	Scheduled 11/20	D3R26
D2R27	Scheduled 11/21				Scheduled 11/22	D3R27

Note 1: The Unit 3 Third ISI Interval was extended by 80 days as permitted by Paragraph IWA-2430(d). This extension is being carried forward to the Fifth ISI Interval to accommodate both Units 2 and 3 having the same interval start date. No interval overlap is being implemented, and thus the start dates for the Fifth ISI Interval are being adjusted the same. As required by Paragraph IWA-2430(c)(1), successive intervals shall not be altered by more than one year from the original pattern. This means that for the remainder of the Fifth ISI Interval, only 285 days are available to use under the Paragraph IWA-2430(c) extension.

Note 2: The Unit 2 second period of the Fifth ISI Interval was reduced by 110 days as permitted by Paragraph IWA-2430(c)(3) to include D2R26 in the third period since it was scheduled to start in November 2019. (Note that the Unit 3 second period of the Fifth ISI Interval was also extended by 110 days as permitted by Paragraph IWA-2430(c)(3) for unit consistency.)

TABLE 1.1-2
UNITS 2 & 3 SECOND CISI INTERVAL/PERIOD/OUTAGE MATRIX
(FOR ISI CLASS MC COMPONENT EXAMINATIONS)

Unit 2		Period	Interval	Period	Unit 3	
Outage Number	**Projected Outage Start Date or Outage Duration**	**Start Date to End Date**	**Start Date to End Date**	**Start Date to End Date**	**Projected Outage Start Date or Outage Duration**	**Outage Number**
D2R21	Scheduled 10/09	1st 9/9/08 to 9/8/11	2nd (Unit 2) 9/9/08 to 9/8/18 2nd (Unit 3) 9/9/08 to 9/8/18	1st 9/9/08 to 9/8/11	Scheduled 10/08	D3R20
D2R22	Scheduled 10/11	2nd 9/9/11 to 9/8/15			Scheduled 10/10	D3R21
D2R23	Scheduled 10/13			2nd 9/9/11 to 9/8/15	Scheduled 10/12	D3R22
D2R24	Scheduled 10/15	3rd 9/9/15 to 9/8/18			Scheduled 10/14	D3R23
D2R25	Scheduled 10/17			3rd 9/9/15 to 9/8/18	Scheduled 10/16	D3R24

1.2 Background

The Commonwealth Edison Company, now known commercially as Exelon Generation Company (Exelon), obtained construction permits to build DNPS on January 10, 1966, for Unit 2, CPPR-18, and on October 14, 1966, for Unit 3, CPPR-22. The docket numbers assigned to DNPS are 50-237 for Unit 2 and 50-249 for Unit 3. After satisfactory plant construction and preoperational testing was completed, DNPS was granted a full power operating license for Unit 2, DPR-19, and subsequently commenced commercial operation on June 9, 1970; the full power operating license for Unit 3, DPR-25, was granted and commercial operation commenced on November 16, 1971.

DNPS's piping systems and associated components were designed and fabricated before the inspection and testing requirements of ASME Section XI were formalized and published. Since this plant was not specifically designed to meet the inspection and testing requirements of ASME Section XI, literal compliance is not feasible or practical within the limits of the current plant design. Certain limitations are likely to occur due to conditions such as accessibility, geometric configuration, and/or metallurgical characteristics. For some inspection categories, an alternate component may be selected for examination and the code statistical and distribution requirements can still be maintained. If ASME Section XI required examination criteria cannot be met, a relief request will be submitted in accordance with 10CFR50.55a.

1.3 Third Interval ISI Program

Pursuant to the Code Of Federal Regulations, Title 10, Part 50, Section 55a, *Codes and standards*, (10CFR50.55a), Paragraph (g), *Inservice inspection requirements*, DNPS was required to update the ISI Program to meet the requirements of ASME Section XI once every ten years or inspection interval. The ISI Program was required to comply with the latest Edition and Addenda of ASME Section XI incorporated by reference in 10CFR50.55a twelve months prior to the start of the interval per 10CFR50.55a(g)(4)(ii). Accordingly, the Inservice Inspection requirements applicable to the Third Interval ISI Program should have been based on the rules set forth in the 1986 Edition, No Addenda of ASME Section XI.

However, ComEd by letter dated April 26, 1991, requested NRC approval to meet the requirements set forth in the 1989 Edition, 1989 Addenda of ASME Section XI prior to its incorporation by reference into 10CFR50.55a(b)(2). NRC approval was received to utilize the 1989 Edition, No Addenda under the letter from R. J. Barrett to T. J. Kovach dated January 28, 1992, "Inservice Inspection Program Update - Dresden, Units 2 and 3." Therefore, the DNPS Third Interval ISI Program Plan was developed in accordance with the requirements of the 1989

Edition, No Addenda of ASME Section XI. The ISI Program Plan addressed Subsections IWA, IWB, IWC, IWD, and IWF of ASME Section XI.

DNPS adopted the EPRI Topical Report TR-112657, Rev. B-A methodology, which was supplemented by ASME Code Case N-578-1, for implementing risk-informed inservice inspections. The RISI Program was in effect from the middle of the third period through the end of the Third ISI Interval. This approach replaced the categorization, selection, and examination volume requirements of ASME Section XI Examination Categories B-F, B-J, C-F-1, and C-F-2 applicable to DNPS with Examination Category R-A as defined in ASME Code Case N-578-1.

The Unit 2 Third ISI Interval was extended by 325 days as permitted by Paragraph IWA-2430(e) and successive inspection interval and periods were adjusted accordingly. The Unit 3 Third ISI Interval was extended by 80 days as permitted by Paragraph IWA-2430(d). These extensions were carried forward to the Fourth ISI Interval to accommodate both Units 2 and 3 having the same interval start date.

Therefore, the DNPS Third ISI Interval was effective from March 1, 1992, through January 19, 2003, for Units 2 and 3, respectively.

1.4 Fourth Interval ISI Program

Pursuant to 10CFR50.55a(g), DNPS was required to update the ISI Program to meet the requirements of ASME Section XI once every ten years or inspection interval. The ISI Program was required to comply with the latest Edition and Addenda of the Code incorporated by reference in 10CFR50.55a twelve months prior to the start of the Fourth ISI Interval per 10CFR50.55a(g)(4)(ii).

The DNPS Fourth ISI Interval started on January 20, 2003, for Units 2 and 3 using the requirements of 10CFR50.55a, and the 1995 Edition through the 1996 Addenda of ASME Section XI. The DNPS Fourth Interval ISI Program Plan addressed Subsections IWA, IWB, IWC, IWD, IWF, Mandatory Appendices of ASME Section XI, approved Code Cases, approved alternatives through relief requests and Safety Evaluations (SE’s).

The Unit 3 Third ISI Interval was extended by 80 days as permitted by Paragraph IWA-2430(d). This extension is being carried forward to the Fifth ISI Interval to accommodate both Units 2 and 3 having the same interval start date. No interval overlap is being implemented, and thus the start dates for the Fifth ISI Interval are being adjusted the same.

The DNPS Fourth ISI Interval was effective from January 20, 2003, through January 19, 2013, for Units 2 and 3, respectively.

1.5 Fifth Interval ISI Program

Pursuant to 10CFR50.55a(g), licensees are required to update their ISI Programs to meet the requirements of ASME Section XI once every ten years or inspection interval. The ISI Program is required to comply with the latest Edition and Addenda of the Code incorporated by reference in 10CFR50.55a twelve months prior to the start of the Fifth ISI Interval per 10CFR50.55a(g)(4)(ii). As discussed in Section 1.4 above, the start of the Fifth ISI Interval will be on January 20, 2013, for DNPS Units 2 and 3. Based on this date, the latest Edition and Addenda of the Code referenced in 10CFR50.55a(b)(2) twelve months prior to the start of the Fifth ISI Interval was the 2007 Edition through the 2008 Addenda.

The DNPS Fifth Interval ISI Program Plan was developed in accordance with the requirements of 10CFR50.55a, and the 2007 Edition through the 2008 Addenda of ASME Section XI, subject to the limitations and modifications contained within Paragraph (b) of the regulation. These limitations and modifications are detailed in Table 1.8-1 of this section. This ISI Program Plan addresses Subsections IWA, IWB, IWC, IWD, IWF, Mandatory Appendices of ASME Section XI, approved Code Cases, approved alternatives through relief requests and SE's, and utilizes the Inspection Program.

DNPS adopted the EPRI Topical Report TR-112657, Rev. B-A methodology, which was supplemented by ASME Code Case N-578-1, for implementing risk-informed inservice inspections during the Third ISI Interval. The RISI Program will continue for the Fifth ISI Interval. Implementation of the RISI Program is in accordance with Relief Request I5R-02.

The DNPS Fifth ISI Interval is effective from January 20, 2013, through January 19, 2023, for Units 2 and 3, respectively.

1.6 First Interval CISI Program

CISI examinations were originally invoked by amended regulations contained within a Final Rule issued by the NRC. The amended regulation incorporated the requirements of the 1992 Edition through the 1992 Addenda of the ASME Section XI, Subsection IWE, subject to specific modifications that were included in Paragraph 10CFR50.55a(b)(2)(x).

The final rulemaking was published in the Federal Register on August 8, 1996, and specified an effective date of September 9, 1996. Implementation of the Subsection IWE Program from a scheduling standpoint was driven by the five year expedited implementation period per 10CFR50.55a(g)(6)(ii)(B), which specified that the examinations required to be completed by the end of the first period of the First CISI Interval (per Table IWE-2412-1) be completed by the effective date (by September 9, 2001).

ASME Section XI Subsections IWE, Mandatory Appendices of ASME Section XI, approved IWE Code Cases, and approved alternatives through relief requests and SE's were developed to implement these requirements.

The DNPS First CISI Interval was effective from September 9, 1996, through September 8, 2008, for Units 2 and 3, respectively.

A request for schedule exemption was submitted to extend the five year period for expedited examination of containment for a maximum of 90 days to allow completion of the required first period examinations during D2R17 in accordance with Relief Request MCR-02, which also implemented the 1998 Edition, No Addenda of ASME Section XI. As permitted by Paragraph IWA-2430(d)(3), a one year extension was also taken during the second period to allow completion of the required Second Period examinations for D2R18. These extensions did not affect the start of the Second CISI Interval.

1.7 Second Interval CISI Program

Pursuant to 10CFR50.55a(g), licensees are required to update their CISI Programs to meet the requirements of ASME Section XI once every ten years or inspection interval. The CISI Program is required to comply with the latest Edition and Addenda of ASME Section XI incorporated by reference in 10CFR50.55a twelve months prior to the start of the interval per 10CFR50.55a(g)(4)(ii). The start of the Second CISI Interval was September 9, 2008. Based on this date, the latest Edition and Addenda of ASME Section XI referenced in 10CFR50.55a(b)(2) twelve months prior to the start of the Second CISI Interval was the 2001 Edition through the 2003 Addenda.

The DNPS Second Interval CISI Program Plan was developed in accordance with the requirements of 10CFR50.55a and the 2001 Edition through the 2003 Addenda of ASME Section XI, subject to the limitations and modifications contained within Paragraph (b) of the regulation. These limitations and modifications are detailed in Table 1.8-1 of this section. This Second Interval CISI Program Plan addresses Subsections IWE, Mandatory Appendices of ASME Section XI, approved IWE Code Cases, approved alternatives through relief requests and SE's, and utilizes Inspection Program B.

The DNPS Second CISI Interval is effective from September 9, 2008, through September 8, 2018, for Units 2 and 3, respectively.

1.8 Code of Federal Regulations 10CFR50.55a Requirements

There are certain Paragraphs in 10CFR50.55a that list the limitations, modifications, and/or clarifications to the implementation requirements of ASME Section XI. These Paragraphs in 10CFR50.55a that are applicable to DNPS are detailed in Table 1.8-1.

TABLE 1.8-1
CODE OF FEDERAL REGULATIONS 10CFR50.55a REQUIREMENTS

10CFR50.55a Paragraphs	Limitations, Modifications, and Clarifications
10CFR50.55a(b)(2)(ix)(A)	**(CISI)** *Examination of metal containments and the liners of concrete containments:* For Class MC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall provide the following in the ISI Summary Report as required by IWA-6000: (*1*) A description of the type and estimated extent of degradation, and the conditions that led to the degradation; (*2*) An evaluation of each area, and the result of the evaluation, and; (*3*) A description of necessary corrective actions.
10CFR50.55a(b)(2)(ix)(B)	**(CISI)** *Examination of metal containments and the liners of concrete containments:* When performing remotely the visual examinations required by Subsection IWE, the maximum direct examination distance specified in Table IWA-2210-1 may be extended and the minimum illumination requirements specified in Table IWA-2210-1 may be decreased, provided that the conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination.
10CFR50.55a(b)(2)(ix)(F)	**(CISI)** *Examination of metal containments and the liners of concrete containments:* VT-1 and VT-3 examinations must be conducted in accordance with IWA-2200. Personnel conducting examinations in accordance with the VT-1 or VT-3 examination method shall be qualified in accordance with IWA-2300. The "owner-defined" personnel qualification provisions in IWE-2330(a) for personnel that conduct VT-1 and VT-3 examinations are not approved for use.

TABLE 1.8-1
CODE OF FEDERAL REGULATIONS 10CFR50.55a REQUIREMENTS

10CFR50.55a Paragraphs	Limitations, Modifications, and Clarifications
10CFR50.55a(b)(2)(ix)(G)	**(CISI)** *Examination of metal containments and the liners of concrete containments:* The VT-3 examination method must be used to conduct the examinations in Items E1.12 and E1.20 of Table IWE-2500-1, and the VT-1 examination method must be used to conduct the examination in Item E4.11 of Table IWE-2500-1. An examination of the pressure-retaining bolted connections in Item E1.11 of Table IWE-2500-1 using the VT-3 examination method must be conducted once each interval. The "owner-defined" visual examination provisions in IWE-2310(a) are not approved for use for VT-1 and VT-3 examinations.
10CFR50.55a(b)(2)(ix)(H)	**(CISI)** *Examination of metal containments and the liners of concrete containments:* Containment bolted connections that are disassembled during the scheduled performance of the examinations in Item E1.11 of Table IWE-2500-1 must be examined using the VT-3 examination method. Flaws or degradation identified during the performance of a VT-3 examination must be examined in accordance with the VT-1 examination method. The criteria in the material specification or IWB-3517.1 must be used to evaluate containment bolting flaws or degradation. As an alternative to performing VT-3 examinations of containment bolted connections that are disassembled during the scheduled performance of Item E1.11, VT-3 examinations of containment bolted connections may be conducted whenever containment bolted connections are disassembled for any reason.
10CFR50.55a(b)(2)(ix)(I)	**(CISI)** *Examination of metal containments and the liners of concrete containments:* The ultrasonic examination acceptance standard specified in IWE-3511.3 for Class MC pressure-retaining components must also be applied to metallic liners of Class CC pressure-retaining components.
10CFR50.55a(b)(2)(xii)	**(ISI)** *Underwater welding:* The provisions in IWA-4660, "Underwater Welding," of Section XI, 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, are not approved for use on irradiated material.

TABLE 1.8-1
CODE OF FEDERAL REGULATIONS 10CFR50.55a REQUIREMENTS

10CFR50.55a Paragraphs	Limitations, Modifications, and Clarifications
10CFR50.55a(b)(2)(xviii)(A)	**(ISI)** *Certification of NDE personnel:* Level I and II nondestructive examination personnel shall be recertified on a 3-year interval in lieu of the 5-year interval specified in the 1997 Addenda and 1998 Edition of IWA-2314, and IWA-2314(a) and IWA-2314(b) of the 1999 Addenda through the latest Edition and Addenda incorporated by reference in paragraph (b)(2) of this section.
10CFR50.55a(b)(2)(xix)	**(ISI)** *Substitution of alternative methods:* The provisions for substituting alternative examination methods, a combination of methods, or newly developed techniques in the 1997 Addenda of IWA-2240 must be applied when using the 1998 Edition through the 2004 Edition of Section XI of the ASME B&PV Code. The provisions in IWA-4520(c), 1997 Addenda through the 2004 Edition, allowing the substitution of alternative methods, a combination of methods, or newly developed techniques for the methods specified in the Construction Code are not approved for use. The provisions in IWA-4520(b)(2) and IWA-4521 of the 2008 Addenda through the latest edition and addenda approved in paragraph (b)(2) of this section, allowing the substitution of ultrasonic examination for radiographic examination specified in the Construction Code are not approved for use.
10CFR50.55a(b)(2)(xx)(B)	**(ISI)** *System leakage tests:* The NDE provision in IWA-4540(a)(2) of the 2002 Addenda of Section XI must be applied when performing system leakage tests after repair and replacement activities performed by welding or brazing on a pressure retaining boundary using the 2003 Addenda through the latest Edition and Addenda incorporated by reference in paragraph (b)(2) of this section.
10CFR50.55a(b)(2)(xxii)	**(ISI)** *Surface Examination:* The use of the provision in IWA-2220, “Surface Examination,” of Section XI, 2001 Edition through the latest Edition and Addenda incorporated by reference in paragraph (b)(2) of this section, that allow use of an ultrasonic examination method is prohibited.
10CFR50.55a(b)(2)(xxiii)	**(ISI)** *Evaluation of Thermally Cut Surfaces:* The use of the provisions for eliminating mechanical processing of thermally cut surfaces in IWA-4461.4.2 of Section XI, 2001 Edition through the latest Edition and Addenda incorporated by reference in Paragraph (b)(2) of this section are prohibited.

TABLE 1.8-1
CODE OF FEDERAL REGULATIONS 10CFR50.55a REQUIREMENTS

10CFR50.55a Paragraphs	Limitations, Modifications, and Clarifications
10CFR50.55a(b)(2)(xxiv)	**(PDI)** *Incorporation of the performance demonstration initiative and addition of ultrasonic examination criteria:* The use of Appendix VIII and the supplements to Appendix VIII and Article I-3000 of Section XI of the ASME B&PV Code, 2002 Addenda through the 2006 Addenda is prohibited.
10CFR50.55a(b)(2)(xxv)	**(ISI)** *Mitigation of Defects by Modification:* The use of the provisions in IWA-4340, “Mitigation of Defects by Modification,” Section XI, 2001 Edition through the latest Edition and Addenda incorporated by reference in Paragraph (b)(2) of this section are prohibited.
10CFR50.55a(b)(2)(xxvi)	**(SPT)** *Pressure Testing Class 1, 2, and 3 Mechanical Joints:* The repair and replacement activity provisions in IWA-4540(c) of the 1998 Edition of Section XI for pressure testing Class 1, 2, and 3 mechanical joints must be applied when using the 2001 Edition through the latest Edition and Addenda incorporated by reference in Paragraph (b)(2) of this section.
10CFR50.55a(b)(2)(xxvii)	**(ISI)** *Removal of Insulation:* When performing visual examinations in accordance with IWA-5242 of Section XI of the ASME B&PV Code, 2003 Addenda through the 2006 Addenda, or IWA-5241 of the 2007 Edition through the latest Edition and Addenda incorporated in Paragraph (b)(2) of this section, insulation must be removed from 17-4 PH or 410 stainless steel studs or bolts aged at a temperature below 1100 °F or having a Rockwell Method C hardness value above 30, and from A-286 stainless steel studs or bolts preloaded to 100,000 pounds per square inch or higher.
10CFR50.55a(b)(2)(xxix)	**(ISI)** *Nonmandatory Appendix R:* Nonmandatory Appendix R, “Risk-Informed Inspection Requirements for Piping,” of Section XI, 2005 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, may not be implemented without prior NRC authorization of the proposed alternative in accordance with paragraph (a)(3)(i) of this section.

TABLE 1.8-1
CODE OF FEDERAL REGULATIONS 10CFR50.55a REQUIREMENTS

10CFR50.55a Paragraphs	Limitations, Modifications, and Clarifications
10CFR50.55a(b)(3)(v)(B)	**(ISI)** *Subsection ISTD:* Article IWF-5000, "Inservice Inspection Requirements for Snubbers," of the ASME B&PV Code, Section XI, must be used when performing inservice inspection examinations and tests of snubbers at nuclear power plants, except as conditioned in Paragraph (b)(3)(v)(B) of this section. (B) Licensees shall comply with the provisions for examining and testing snubbers in Subsection ISTD of the ASME OM Code and make appropriate changes to their technical specifications or licensee-controlled documents when using the 2006 Addenda and later editions and addenda of Section XI of the ASME B&PV Code.
10CFR50.55a(b)(5)	**(ISI)** *Inservice Inspection Code Cases:* Licensees may apply the ASME Boiler and Pressure Vessel Code Cases listed in Regulatory Guide 1.147, Revision 16, without prior NRC approval subject to the following: (i) When a licensee initially applies a listed Code Case, the licensee shall apply the most recent version of that Code Case incorporated by reference in this paragraph. (ii) If a licensee has previously applied a Code Case and a later version of the Code Case is incorporated by reference in this paragraph, the licensee may continue to apply, to the end of the current 120-month interval, the previous version of the Code Case as authorized or may apply the later version of the Code Case, including any NRC-specified conditions placed on its use. (iii) Application of an annulled Code Case is prohibited unless a licensee previously applied the listed Code Case prior to it being listed as annulled in Regulatory Guide 1.147. Any Code Case listed as annulled in any Revision of Regulatory Guide 1.147 which a licensee has applied prior to it being listed as annulled, may continue to be applied by that licensee to the end of the 120-month interval in which the Code Case was implemented.

TABLE 1.8-1
CODE OF FEDERAL REGULATIONS 10CFR50.55a REQUIREMENTS

10CFR50.55a Paragraphs	Limitations, Modifications, and Clarifications
10CFR50.55a(b)(6)	**(ISI)** *Operation and Maintenance of Nuclear Power Plants Code Cases:* Licensees may apply the ASME Operation and Maintenance Nuclear Power Plants Code Cases listed in Regulatory Guide 1.192 without prior NRC approval subject to the following: (i) When a licensee initially applies a listed Code Case, the licensee shall apply the most recent version of that Code Case incorporated by reference in this paragraph. (ii) If a licensee has previously applied a Code Case and a later version of the Code Case is incorporated by reference in this paragraph, the licensee may continue to apply, to the end of the current 120-month interval, the previous version of the Code Case as authorized or may apply the later version of the Code Case, including any NRC-specified conditions placed on its use. (iii) Application of an annulled Code Case is prohibited unless a licensee previously applied the listed Code Case prior to it being listed as annulled in Regulatory Guide 1.192. If a licensee has applied a listed Code Case that is later listed as annulled in Regulatory Guide 1.192, the licensee may continue to apply the Code Case to the end of the current 120-month interval.

1.9 Code Cases

Per 10CFR50.55a(b)(5), Code Cases that have been determined to be suitable for use in ISI Program Plans by the NRC are listed in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability-ASME Section XI, Division 1." The approved Code Cases in Regulatory Guide 1.147 being utilized by DNPS are included in Section 2.1.1 of this document. The most recent version of a given Code Case incorporated in the revision of Regulatory Guide 1.147, referenced in 10CFR50.55a(b)(5)(i), at the time it is applied within the ISI Program shall be used. The latest version of Regulatory Guide 1.147 incorporated into this document is Revision 16. As this guide is revised, newly approved Code Cases will be assessed for plan implementation at DNPS per Paragraph IWA-2441(e) and proposed for use in revisions to the ISI Program Plan.

The use of other Code Cases (than those listed in Regulatory Guide 1.147) may be authorized by the Director of the Office of Nuclear Reactor Regulation upon request pursuant to 10CFR50.55a(a)(3). Code Cases not approved for use in Regulatory Guide 1.147, which are being utilized by DNPS through associated relief requests that are included in Sections 8.0.

Per 10CFR50.55a(b)(6), this ISI Program Plan will also utilize Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code." The approved Code Cases in Regulatory Guide 1.192, which are being utilized by DNPS, are included in Section 2.1.2. The latest version of Regulatory Guide 1.192 incorporated into this document is Revision 0. As this guide is revised, newly approved Code Cases will be assessed for plan implementation at DNPS per Paragraph IWA-2441(e) and proposed for use in revisions to the ISI Program Plan.

1.10 Relief Requests

In accordance with 10CFR50.55a, when a licensee either proposes alternatives to ASME Section XI requirements which provide an acceptable level of quality and safety, determines compliance with ASME Section XI requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, or determines that specific ASME Section XI requirements for inservice inspection are impractical, the licensee shall notify the NRC and submit information to support the determination.

The submittal of this information will be referred to in this document as a "relief request." Relief requests for the Fifth ISI Interval and the Second CISI Interval are included in Section 8.0 of this document. The text of the relief requests contained in Section 8.0 will demonstrate one of the following: the proposed alternatives provide an acceptable level of quality and safety per 10CFR50.55a(a)(3)(i), compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety per 10CFR50.55a(a)(3)(ii), or the code requirements are considered impractical per 10CFR50.55a(g)(5)(iii).

Per 10CFR50.55a Paragraphs (a)(3) and (g)(6)(i), the Director of the Office of Nuclear Reactor Regulation will evaluate relief requests and "may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility."

2.0 BASIS FOR INSERVICE INSPECTION PROGRAM

2.1 ASME Section XI Examination Requirements

As required by 10CFR50.55a, this program was developed in accordance with the requirements detailed in the 2007 Edition through the 2008 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsections IWA, IWB, IWC, IWD, IWE, IWF, Mandatory Appendices, Inspection Program of Paragraph IWA-2431, and approved alternatives through relief requests and SE's.

The Performance Demonstration Initiative (PDI) is an organization comprised of all US nuclear utilities that was formed to provide an efficient implementation of Appendix VIII performance demonstration requirements. The Electric Power Research Institute (EPRI) NDE Center was selected as the administrator of this program. The PDI program is administered according to the "PDI Program Description." The ISI Program implements Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," ASME Section XI 2007 Edition through the 2008 Addenda as required by 10CFR50.55a(b)(2)(xxiv) and with modifications as identified in 10CFR50.55a(b)(2)(xiv), (xv), and (xvi). Appendix VIII requires qualification of the procedures, personnel, and equipment used to detect and size flaws in piping, bolting, and the reactor pressure vessel (RPV). Each organization (e.g., owner or vendor) will be required to have a written program to ensure compliance with the requirements. DNPS maintains the responsibility to ensure that Appendix VIII requirements are properly implemented.

For the Fifth ISI Interval, DNPS's inspection program for ASME Section XI Examination Categories B-F, B-J, C-F-1, and C-F-2 will be governed by risk-informed requirements. The RISI Program methodology is described in the EPRI Topical Report TR-112657, Rev. B-A. To supplement the EPRI Topical Report, ASME Code Case N-578-1 (as applicable per Relief Request I5R-02) is also being used for the classification of piping structural elements under the RISI Program. The RISI Program scope has been implemented as an alternative to the 2007 Edition through the 2008 Addenda, ASME Section XI examination program for ISI Class 1 B-F and B-J welds and ISI Class 2 C-F-1 and C-F-2 welds in accordance with 10CFR50.55a(a)(3)(i). The basis for the resulting risk categorizations of the nonexempt ISI Class 1 and 2 piping systems at DNPS is defined and maintained in the Final Report, "Risk Informed Inservice Inspection Evaluation," as referenced in Section 9.0 of this document. References to ASME Section XI Examination Categories B-F, B-J, C-F-1, and C-F-2 have been replaced with Examination Category R-A to identify them as part of the RISI Program.

The CISI Program Plan per Subsection IWE has been incorporated into Section 6.0 "Containment ISI Plan" of this ISI Program Plan. The CISI relief requests are included in Section 8.0 of this document.

2.1.1 ASME Section XI Code Cases

As referenced by 10CFR50.55a(b)(5) and allowed by NRC Regulatory Guide 1.147, Revision 16, the following Code Cases are being incorporated into the DNPS ISI Program.

N-62-7 Internal and External Valve Items, Section III, Division 1, Classes 1, 2, and 3

Code Case N-62-7 is acceptable subject to the following conditions in Regulatory Guide 1.84, Revision 35:

The Code requires that ISI Class 1 and 2 valve manufacturers meet the provisions of NCA-4000, "Quality Assurance." ISI Class 3 valve manufacturers must also meet the provisions of NCA-4000 because all Code Class valve items are subject to the licensee's 10CFR, Part 50, Appendix B approved QA program.

Note: Code Case N-62-7 (approved for use per Regulatory Guide 1.84, Revision 35) was utilized for guidance in defining pressure retaining bolting for valves.

N-405-1 Socket Welds, Section III, Division 1

Note: Code Case N-405-1 (approved for use per Regulatory Guide 1.84, Revision 35) was utilized for guidance for appurtenances (with outside diameter equal to that of 2 in. standard pipe size and less) that connect to nozzles of a Section III, Class 1 vessels. These appurtenances may be designed and constructed using weld joints in accordance with Fig. 1 and provided that requirements (a through e) listed in the Code Case are met.

N-432-1 Repair Welding Using Automatic or Machine Gas Tungsten-Arc Welding (GTAW) Temper Bead Technique. Regulatory Guide 1.147, Revision 16.

N-513-3 Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping.

Code Case N-513-3 is acceptable subject to the following condition specified in Regulatory Guide 1.147, Revision 16:

The repair or replacement activity temporarily deferred under the provisions of this Code Case shall be performed during the next scheduled outage.

N-516-3 Underwater Welding

Code Case N-516-3 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 16.

Licensees must obtain NRC approval in accordance with 10CFR50.55a(a)(3) regarding the technique to be used in the weld repair or replacement of irradiated material underwater.

N-526 Alternative Requirements for Successive Inspections of Class 1 and 2 Vessels. Regulatory Guide 1.147, Revision 16.

N-532-4 Alternative Requirements to Repair and Replacement Documentation Requirements and Inservice Summary Report Preparation and Submission as Required by IWA-4000 and IWA-6000. Regulatory Guide 1.147, Revision 16.

Note: Limited to the 2005 Addenda by references in Table 3 of the Code Case. Code Case N-532-4 to be used in accordance with Relief Request I5R-10.

N-578-1 Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B.

Note: Code Case N-578-1 to be used in accordance with Relief Request I5R-02.

N-586-1 Alternative Additional Examination Requirements for Class 1, 2, and 3 Piping, Components, and Supports. Regulatory Guide 1.147, Revision 16.

Note: RISI Program Relief Request I5R-02 requires that scope expansion for RISI elements will be determined using Paragraph -2430 of ASME Code Case N-578-1.

N-597-2 Requirements for Analytical Evaluation of Pipe Wall Thinning

Code Case N-597-2 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 16.

(1) Code Case must be supplemented by the provisions of EPRI Nuclear Safety Analysis Center Report 202L-R2, April 1999, "Recommendations for an Effective Flow Accelerated Corrosion Program," for developing the inspection requirements, the method of predicting the rate of wall thickness loss, and the value of the predicted remaining wall thickness. As used in NSAC-202L-R2, the term "should" is to be applied as "shall" (i.e., a requirement).
(2) Components affected by flow-accelerated corrosion to which this Code Case are applied must be repaired or replaced in accordance with the construction code of record and Owner's requirements or a later NRC approved edition of Section III, "Rules for Construction of Nuclear Plant Components," of the ASME Code prior to the value of t_p reaching the allowable minimum wall thickness, t_{min}, as specified in -3622.1(a)(1) of this Code Case. Alternatively, use of the Code Case is subject to NRC review and approval per 10CFR50.55a(a)(3).
(3) For Class 1 piping not meeting the criteria of -3221, the use of evaluation methods and criteria is subject to NRC review and approval per 10CFR50.55a(a)(3).
(4) For those components that do not require immediate repair or replacement, the rate of wall thickness loss is to be used to determine a suitable inspection frequency so that repair or replacement occurs prior to reaching allowable minimum wall thickness, t_{min}.
(5) For corrosion phenomenon other than flow accelerated corrosion, use of the Code Case is subject to NRC review and approval per 10CFR50.55a(a)(3). Inspection plans and wall thinning rates may be difficult to justify for certain degradation mechanisms such as MIC and pitting.

N-600 Transfer of Welder, Welding Operator, Brazer, and Brazing Operator Qualifications Between Owners. Regulatory Guide 1.147, Revision 16.

N-606-1 Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique for BWR CRD Housing/Stud Tube Repairs

Code Case N-606-1 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 16:

> Prior to welding, an examination or verification must be performed to ensure proper preparation of the base metal, and that the surface is properly contoured so that an acceptable weld can be produced. The surfaces to be welded, and surfaces adjacent to the weld, are to be free from contaminants, such as, rust, moisture, grease, and other foreign material or any other condition that would prevent proper welding and adversely affect the quality or strength of the weld. This verification is to be required in the welding procedures.

N-613-1 Ultrasonic Examination of Full Penetration Nozzles in Vessels, Examination Category B-D, Item Nos. B3.10 and B3.90, Reactor Nozzle-to-Vessel Welds, Figs. IWB-2500-7(a), (b), and (c). Regulatory Guide 1.147, Revision 16.

N-629 Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials. Regulatory Guide 1.147, Revision 16.

N-638-4 Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique

Code Case N-638-4 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 16:

(1) Demonstration for ultrasonic examination of the repaired volume is required using representative samples which contain construction type flaws.
(2) The provisions of 3(e)(2) or 3(e)(3) may only be used when it is impractical to use the interpass temperature measurement methods described in 3(e)(1), such as in situations where the weldment area is inaccessible (e.g., internal bore welding) or when there are extenuating radiological conditions.

Note: Limited to the 2004 Edition by references in Table 1 of the Code Case.

N-639 Alternative Calibration Block Material

Code Case N-639 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 16:

> Chemical ranges of the calibration block may vary from the materials specification if (1) it is within the chemical range of the component specification to be inspected, and (2) the phase and grain shape are maintained in the same ranges produced by the thermal process required by the material specification.

N-641 Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements. Regulatory Guide 1.147, Revision 16.

N-649 Alternative Requirements for IWE-5240 Visual Examination. Regulatory Guide 1.147, Revision 16.

Note: Limited to the 2000 Addenda by references in the Code Case.

N-651 Ferritic and Dissimilar Metal Welding Using SMAW Temper Bead Technique Without Removing the Weld Bead Crown of the First Layer. Regulatory Guide 1.147, Revision 16.

N-661-1 Alternative Requirements for Wall Thickness Restoration of Class 2 and 3 Carbon Steel Piping for Raw Water Service

Code Case N-661-1 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 16.

(1) If the cause of the degradation has not been determined, the repair is only acceptable until the next refueling outage.
(2) When through-wall repairs are made by welding on surfaces that are wet or exposed to water, the weld overlay repair is only acceptable until the next refueling outage.

Note: Limited to the 2005 Addenda in ASME Code Case Applicability Index. Code Case N-661-1 to be used in accordance with Relief Request I5R-11.

N-666 Weld Overlay of Class 1, 2, and 3 Socket Welded Connections. Regulatory Guide 1.147, Revision 16.

Note: Limited to the 2004 Edition by references in Table 1 of the Code Case.

N-705 Evaluation Criteria for Temporary Acceptance of Degradation in Moderate Energy Class 2 or 3 Vessels and Tanks. Regulatory Guide 1.147, Revision 16.

N-730 Roll Expansion of Class 1 Control Rod Drive Bottom Head Penetrations in Boiling Water Reactors (BWR). Regulatory Guide 1.147, Revision 16.

N-733 Mitigation of Flaws in NPS 2 (DN 50) and Smaller Nozzles and Nozzle Partial Penetration Welds in Vessels and Piping by Use of a Mechanical Connection Modification. Regulatory Guide 1.147, Revision 16.

Note: Limited to the 2006 Addenda by references to IWA-5242(a) in (h)(2)(d) of the Code Case.

N-735 Successive Inspection of Class 1 and 2 Piping Welds. Regulatory Guide 1.147, Revision 16.

N-751 Pressure Testing of Containment Penetration Piping. Regulatory Guide 1.147, Revision 16.

Code Case N-751 is acceptable subject to the following conditions specified in Regulatory Guide 1.147, Revision 16:

> When a 10CFR Part 50, Appendix J, Type C test is performed as an alternative to the requirements of IWA-4540 (IWA-4700 in the 1989 Edition through the 1995 Edition) during repair and replacement activities, nondestructive examination must be performed in accordance with IWA-4540(a)(2) of the 2002 Addenda of Section XI.

Additional Code Cases may be invoked in the future shall be in accordance with those approved for use in the latest published revision of Regulatory Guide 1.147 or 10CFR50.55a at that time.

2.1.2 ASME OM Code Cases

As referenced by 10CFR50.55a(b)(6) and allowed by NRC Regulatory Guide 1.192, Revision 0, the following Code Cases are being incorporated into the DNPS ISI Program:

OMN-13, Rev. 0	Requirements for Extending Snubber Inservice Visual Examination Interval at LWR Power Plants

Additional Code Cases invoked in the future shall be in accordance with those approved for use in the latest published revision of Regulatory Guide 1.192 or 10CFR50.55a at that time.

2.2 Augmented Inspection Program Requirements

Augmented inspection program requirements are those inspections that are performed above and beyond the requirements of ASME Section XI. Below is a summary of those inspections performed by DNPS that are not specifically addressed by ASME Section XI, or the inspections that will be performed in addition to the requirements of ASME Section XI on a routine basis during the Fifth ISI Interval (Second CISI Interval).

2.2.1 Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," Revision 2 / Supplement 1 to Generic Letter 88-01, and NUREG-0313, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping," Revision 2, EPRI Topical Report TR-113932, "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)," as conditionally approved by NRC Final SE's dated September 15, 2000, and May 14, 2002, and EPRI Topical Report TR-1012621, "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75-A)," as conditionally approved by NRC final SE dated March 16, 2006

These documents discuss the examination requirements for Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping. References to Generic Letter 88-01 (GL 88-01) within the ISI Program refer to the comprehensive commitments to all of these documents. The final SE's of BWRVIP-75 and BWRVIP-75-A revised the GL 88-01 inspection schedules. The BWRVIP-75 and BWRVIP-75-A revised inspection schedules were based on consideration of inspection results and service experience gained by the industry since issuance of GL 88-01 and NUREG-0313, and includes additional knowledge regarding the benefits of improved BWR water chemistry.

DNPS has committed to the requirements of these documents as discussed in Updated Final Safety Analysis Report (UFSAR) Section 5.2.3.5. The original DNPS commitment concerning Generic Letter 88-01 was sent to the NRC in a letter from W. E. Morgan (CECo) to the NRC dated July 29, 1988. The NRC reviewed this commitment in letters from T. M. Ross (NRC) to T. J. Kovach (CECo) dated May 22, 1989, and from B. L. Siegel (NRC) to T. J. Kovach (CECo) dated August 23, 1990. Since the issuance of GL 88-01, the BWR Vessel and Internals Project (BWRVIP) has been created. This BWR owners group has worked on the mitigation of IGSCC for BWR internal components. As part of their activities, EPRI Topical Report TR-113932, "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)," dated October 27, 1999, and EPRI Topical Report TR-1012621, "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75-A)," dated October 2005, were submitted to the NRC. Among other issues, this document proposed alternative inspection schedules for IGSCC susceptible welds. Two different inspection schedules were presented; one for plants on Normal Water Chemistry (NWC) and one for plants on effective Hydrogen Water Chemistry (HWC). The HWC schedule may be utilized if the applicable performance criteria are met.

After review of BWRVIP-75 and BWRVIP-75-A, the NRC issued a SE approving the documents with minor changes. (Letter from NRC to Carl Terry, BWRVIP Chairman, Final Safety Evaluation of the "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)," dated May 14, 2002, and letter from NRC to Bill Eaton, BWRVIP Chairman, Final Safety Evaluation of the "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75-A)," dated March 16, 2006.)

Based upon NRC endorsement of BWRVIP-75 and BWRVIP-75-A, the DNPS conformance to GL 88-01 inspection schedules was changed to BWRVIP-75 and BWRVIP-75-A for NWC plants except for Category A welds. (See Risk-Informed Inservice Inspection discussion below and BWRVIP discussion in Section 2.2.4).

The outboard RWCU piping has been excluded from Generic Letter 88-01. The basis for this exclusion is documented in a letter from P. L.

Piet (CECo) to T. E. Murley (NRC) dated August 20, 1993, and in a letter from R. M. Pulsifer (NRC) to D. L. Farrar (ComEd) dated September 22, 1994.

RISI regulations have been invoked for DNPS in this ISI Program Plan. Under these new guidelines, ISI Class 1 and 2 piping structural elements are inspected in accordance with EPRI Topical Report TR-112657, Rev. B-A and ASME Code Case N-578-1. Per this Topical Report and Code Case, welds within the plant that are assigned to IGSCC Categories B through G will continue to meet existing IGSCC schedules, while IGSCC Category A welds have been subsumed into the RISI Program.

The First Ten-Year inspection cycle for implementation of BWRVIP-75-A at DNPS Units 2 and 3 began on September 15, 2000. The Second Ten-Year inspection cycle for implementation of BWRVIP-75-A at DNPS Units 2 and 3 began on September 16, 2010. The IGSCC inspection program has been revised to schedule the required examinations.

Implementation of the DNPS program addressing these documents is included in Section 7.0 of this ISI Program Plan and the associated ISI Database.

2.2.2 Boiling Water Reactor Owners' Group (BWROG) Report GE-NE-523-A71-0594-A, Revision 1, "Alternate BWR Feedwater Nozzle Inspection Requirements, May 2000," as approved by NRC final SE dated March 10, 2000, Boiling Water Reactor Owners' Group (BWROG) Report GE-NE-523-A71-0594, "Alternate BWR Feedwater Nozzle Inspection Requirements, August 1999," as conditionally approved by NRC final SE dated June 5, 1998, and NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," dated November 1980

These documents discuss the current and initial examination requirements for BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking. The alternate approach was developed and submitted to the NRC by the BWROG. The NRC accepted these alternate requirements in a final SE dated March 10, 2000.

DNPS initially committed to the requirements of NUREG-0619 as stated in the Third Interval ISI Program Plan. DNPS previously revised this commitment to utilize the BWROG alternate inspections in a letter from J. M. Heffley (ComEd) to the NRC dated May 21, 1999. Future inspections will comply with BWROG "Alternate BWR Feedwater Nozzle Inspection Requirements," GE-NE-523-A71-0594-A, Revision 1, dated May 2000, as accepted by NRC SE (TAC No. MA6787) dated March 10, 2000.

Implementation of the examination commitments is included in Section 7.0 of this ISI Program Plan and the associated ISI Database.

2.2.3 Inspection of Reactor Head Closure Studs

DNPS is currently committed to inspection of enhanced end shot UT of all 92 reactor head studs every refueling outage.

This commitment originated due to stress corrosion cracking (SCC) discovered on Unit 2. A worst case assumption was made that the cracking was a generic problem for DNPS Units 2 and 3. Therefore, volumetric examinations of all reactor head studs were performed every refueling outage. This commitment is documented in LER 91-002. The commitment has been reduced to examination of Unit 2 reactor head studs since the cracking was contributed to the material hardness unique to Unit 2. This commitment reduction is documented in evaluation DG-00-000012, dated January 10, 2000.

2.2.4 BWR Vessel and Internals Project (BWRVIP)

Increased awareness of the presence of in-vessel component degradation has led to the formation of the BWRVIP. BWRVIP is an association of BWR utilities focused on the common purpose of investigating and developing effective and acceptable approaches for addressing in-vessel component degradation through improved detection, mitigation, and/or repair techniques. In accordance with the BWRVIP charter, the organization is tasked with providing generic resolution to BWR issues and representing the member utilities in negotiating with the NRC for approval of the groups' recommended actions. Exelon, as a member utility of the BWRVIP, has endorsed the objectives prescribed by the BWRVIP.

The BWRVIP is comprised of a series of Inspection & Evaluation Guidelines and documents that discuss RPV internals. The BWRVIP encompasses pertinent information and requirements presented in I.E. Bulletins (IEBs), General Electric (GE) Service Information Letters (SILs), and Rapid Information Communication Services Information Letters (RICSILs). The BWRVIP guidelines are intended to be followed in lieu of GE SILs and RICSILs issued prior to issuance of the BWRVIP guidelines.

Exelon's commitments to the BWRVIP are discussed in BWRVIP letters to the NRC dated May 30, 1997, and October 30, 1997. The NRCs response to the discussion of BWRVIP utility commitments is discussed in an NRC letter to the BWRVIP dated July 29, 1997.

Examinations of RPV internals, as required by DNPS commitments to the BWRVIP and other related documents, are performed in accordance with procedure ER-AB-331.

Implementation of BWRVIP inspection requirements in lieu of ASME Section XI inspection requirements for Examination Categories B-N-1 and B-N-2 was requested in Relief Request I5R-07.

2.2.5 NUREG-1796 Portions Regarding Nonexempt Class MC Piping and Class MC Component Supports, dated October 2004

This commitment results from resolution of Open Item (OI) 3.5.2.3.2-1 in NUREG-1796 "Safety Evaluation Report Related to the License Renewal of the Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2," and the Exelon response letter dated April 22, 2004.

Nonexempt MC Piping Supports

The details of the augmented piping support inspections are contained in License Renewal Application (LRA), Appendix B, Item B.1.30, and in Commitment #30 in the associated SE, as well as in UFSAR Supplement Appendix A (see ATI # 101562.49.48).

The commitment modifies the station's Structural Monitoring Program (SMP) to perform augmented Class MC piping support inspections. These augmented inspections are to include a 15% sample size of nonexempt piping (> 4" NPS) supports, distributed across systems with such supports, and performed by certified VT-3 visual inspectors. The total number of nonexempt Class MC piping supports under this commitment includes 33 Unit 2 and 26 Unit 3 supports from the Reactor Building Closed Cooling Water and Primary Containment Vent and Purge systems.

Class MC Component Supports

The details of the augmented Component Support inspections are contained in License Renewal Application (LRA), Appendix B, Item B.1.27 and the associated SE (see ATI # 101522.37).

The commitment is to perform VT-3 visual examinations on the Drywell, Suppression Chamber, and Vent System supports. These exams will be similar to those required by ASME Section XI, Subsection IWF.

Finally, the NUREG-1796 Piping and Component Support commitment includes performing a baseline inspection of both the sample Class MC piping supports and the Class MC component supports prior to entering the period of extended operation (for Unit 2 - midnight on 12/22/09; and for Unit 3 - midnight on 01/12/11). After these initial baseline inspections have been completed, the 15% piping support sample population and the component supports will be scheduled and inspected during the Fifth ISI Interval and successive intervals thereafter.

Implementation of the examination commitments is included in Sections 6.0 and 7.0 of this ISI Program Plan and the associated ISI Database.

2.2.6 Drywell Shell at the Sand Pocket Location

This DNPS Unit 3 area is inaccessible and is located at the sand pocket which is on the outside of the drywell shell at the bottom of the 2" thermal gap between the drywell shell and the reactor building concrete shield wall. This area was identified as an industry concern in NRC Information Notice 86-99 and Generic Letter 87-05 regarding potential corrosion of the drywell steel shell in the sand pocket region. The concern relates primarily to plants which have their sand pocket regions open to the gap between the drywell and surrounding concrete. Due to their small size, the drains in this region have a greater potential of clogging, and the likelihood of adequately drying the sand is low should copious amounts of water enter the sand cushion region. DNPS has conducted an extensive review of this concern, the results of which are summarized in DNPS UFSAR Section 6.2.1.2.1.2. To date, the concerns raised by the notices are evident but have not resulted in any degradation of the drywell shell. However, since water continues to enter this region, DNPS will consider it as an augmented containment inspection area and it will be inspected as such.

Implementation of the examination commitments is included in Sections 6.0 and 7.0 of this ISI Program Plan and the associated ISI Database.

2.2.7 License Renewal Requirements for Cast Austenitic Stainless Steel Components

The DNPS License Renewal process developed an Aging Management Program (AMP) to address cast austenitic stainless steel (CASS) components in the reactor vessel that are susceptible to a loss of fracture toughness due to thermal aging/neutron embrittlement. This AMP was developed in accordance with NUREG-1801, “Generic Aging Lessons Learned.” Procedure ER-AB-331-101, “Evaluation for Thermal Aging/Neutron Embrittlement of Reactor Internals Components,” identified the following CASS components as susceptible to thermal aging/neutron embrittlement at DNPS: the fuel support piece, the control rod guide tube base, the Jet Pump (JP) mixer flange, JP mixer flare, JP mixer ring, JP inlet-mixer nozzles, and JP inlet-mixer elbows.

The evaluation of the CASS components concluded that periodic inspection was required for each of the identified components. The inspections consist of enhanced VT-1 visual examinations on a sample of the identified components each refueling outage. As part of the license renewal commitment documented in A/R 101522-49-04, these inspections are to be included in the ISI Program. The evaluation and inspection requirements are documented in EC 376712, "Engineering Evaluation for Thermal Aging/Neutron Embrittlement of Reactor Internals Components."

Note: The components and schedule of examinations are maintained and implemented in the DNPS Reactor Internals Program documents.

2.3 System Classifications and P&ID Boundary Drawings

The ISI Classification Basis Document details those systems that are ISI Class 1, 2, or 3 that fall within the inservice inspection scope of examinations including the containment structure (metal). Below is a summary of the classification criteria used within the Basis Document.

Each safety related fluid system containing water, steam, air, oil, etc. included in the DNPS UFSAR was reviewed to determine which safety functions they perform during all modes of system and plant operation. Based on these safety functions, the systems and components were evaluated per classification documents. The systems were then designated as ISI Class 1, 2, 3, or non-classed accordingly. This evaluation followed the guidelines of UFSAR Section 5.2.4 for ISI Class 1 and UFSAR Section 6.6 for ISI Classes 2 and 3. Safety related portions of systems are defined by the Piping and Instrumentation Diagrams (P&IDs) with an "S" flag.

When a particular group of components is identified as performing an ISI Class 1, 2, or 3 safety function, the components are further reviewed to assure the interfaces (boundary valves and boundary barriers) meet the criteria set by 10CFR50.2, 10CFR50.55a(c)(1), 10CFR50.55a(c)(2), and Regulatory Guide 1.26. Although DNPS is not committed to or licensed in accordance with these documents, Standard Review Plan (SRP) 3.2.2, "System Quality Group Classification," and American National Standards Institute/American Nuclear Society (ANSI/ANS) 58.14-1993, "Safety and Pressure Integrity Classification Criteria for Light Water Reactors," were also used for guidance in evaluating the classification boundaries when 10CFR and Regulatory Guide 1.26 did not address a given situation. The valve positions shown on the system flow diagrams are assumed to be the normal positions during system operation unless otherwise noted.

At the time the construction permits for DNPS Units 2 and 3 were issued, ASME Section III covered only pressure vessels, primarily nuclear reactor vessels. The majority of piping, pumps, and valves were designed and installed according to the rules of USAS B31.1.0-1967 Edition, "Power Piping." Consequently, the DNPS ISI Program has essentially no ASME Section III Class 1, 2, or 3 piping systems.

ISI classification boundaries are defined by the P&IDs with a classification flag. The coding designators 1, 2, 3, and MC, respectively, are used for classifying nonexempt ASME Section XI components.

The systems and components (piping, pumps, valves, vessels, etc.), which are subject to the examinations of Articles IWB-2000, IWC-2000, IWD-2000 and IWF-2000, and pressure tests of Articles IWB-5000, IWC-5000, and IWD-5000 are identified on the P&IDs as detailed in Table 2.3-1. The systems and components, which are subject to the examinations of Article IWE-2000 are identified on the P&IDs as detailed in Table 2.3-2.

TABLE 2.3-1
P&ID BOUNDARY DRAWINGS (ISI)

UNIT 2	UNIT 3 & COMMON	TITLE
M-12, SH. 1 & 2	M-345, SH. 1 & 2	Diagram of Main Steam Piping (MS)
M-14	M-347	Diagram of Reactor Feedwater Piping (FW)
M-22	M-355	Diagram of Service Water Piping (CCSW) & (DGSW)
M-25	M-356	Diagram of Pressure Suppression Piping (PS)
M-26, SH. 1, 2 & 3	M-357, SH. 1, 2 & 3	Diagram of Nuclear Boiler & Reactor Recirculation Piping (RR)
M-27	M-358	Diagram of Core Spray Piping (CS)
M-28	M-359	Diagram of Isolation Condenser Piping (ISCO)
M-29, SH. 1 & 2	M-360, SH. 1 & 2	Diagram of L.P. Coolant Injection Piping (LPCI)
M-30	M-361	Diagram of Reactor Water Cleanup Piping (RWCU)
M-32	M-363	Diagram of Shutdown Reactor Cooling Piping (SDC)
M-33	M-364	Diagram of Standby Liquid Control Piping (SBLC)
M-34, SH. 2	M-365, SH. 2	Diagram of Control Rod Drive Hydraulic Piping (CRD)
M-35, SH. 1	M-366	Diagram of Demineralized Water System Piping (DW)
M-39	M-369	Diagram of Reactor Building Equipment Drains
M-51	M-374	Diagram of H.P. Coolant Injection Piping (HPCI)
M-517, SH. 1	M-517, SH. 2 & 3	Diesel Generator Engine Cooling Water System (DGCW)
M-1234, SH. 1	M-1239, SH. 1	P&ID Liquid Sampling
M-3121	M-3121	Diagram of Control Room HVAC

TABLE 2.3-2
P&ID BOUNDARY DRAWINGS (CISI)

UNIT 2	UNIT 3 & COMMON	TITLE
M-20	M-353	Containment - Diagram of Reactor Building Cooling Water Piping (RBCCW)
M-25	M-356	Containment - Diagram of Pressure Suppression Piping (PS)
M-28	M-359	Containment - Diagram of Isolation Condenser Piping (ISCO)
M-29 SH. 1	M-360 SH. 1	Containment - Diagram of L.P. Coolant Injection Piping (LPCI)
M-31	M-362	Containment - Diagram of Fuel Pool Cooling Piping (PC)
M-34 SH. 1	--	Containment - Diagram of Control Rod Drive Hydraulic Piping (CRD)
M-35 SH. 1	M-366	Containment - Diagram of Demineralized Water System Piping (DW)
M-37 SH. 2	M-367 SH. 2 & 3	Containment - Diagram of Instrument Air Piping (IA)
--	M-368	Containment - Diagram of Service Air Piping (SA)
--	M-38 SH. 2	Containment - Diagram of Service Air Piping 2/3 Sparging Air Compressors Reactor Building & Crib House (SA)
M-39	M-369	Containment - Diagram of Reactor Building Equipment Drains
M-178	M-421	Containment - Diagram of Process Sampling Part 2
M-706 SH. 1	M-706 SH. 2	Containment - Diagram of Containment Atmosphere Monitor System (CAM)
M-707 SH. 1	M-707 SH. 2	Containment - Diagram of Atmospheric Containment Atmosphere Dilution System

2.4 ISI Isometric Drawings for Nonexempt ISI Class Components and Supports

ISI isometric drawings were developed to identify the ISI Class 1, 2, and 3 components (welds, bolting, etc.) and support locations at DNPS. SPT isometric drawings were also developed to show those components subject to pressure testing. The ISI isometric and SPT isometric drawings are listed in Table 2.4-1. The ISI Class MC components are identified on the CISI Reference Drawings listed in Tables 2.4-2 and 2.4-3.

DNPS's ISI Program, including the ISI Database, ISI Classification Basis Document, ISI Selection Document and schedule, addresses the nonexempt components which require examination and testing.

A summary of DNPS Units 2 and 3 ASME Section XI nonexempt components and supports is included in Section 7.0.

TABLE 2.4-1
ISI ISOMETRIC AND PRESSURE TESTING DRAWINGS

UNIT 2	UNIT 3 & COMMON	TITLE
ISI-101, SH. 1 & 2	ISI-112, SH. 1 & 2	ISI Class 1 Main Steam Piping
ISI-101, SH. 3	ISI-112, SH. 3	ISI Class 1 Main Steam Drain Piping
ISI-102, SH. 1 & 2	ISI-113, SH. 1 & 2	ISI Class 1 Reactor Feedwater Piping
ISI-103, SH. 1 to 4	ISI-114, SH. 1 to 4	ISI Class 1 Nuclear Boiler and Reactor Recirculation Piping
ISI-104	ISI-115	ISI Class 1 Core Spray Piping
ISI-105	ISI-116	ISI Class 1 Isolation Condenser Piping
ISI-106, SH. 1 & 2	ISI-117, SH. 1 & 2	ISI Class 1 Low Pressure Coolant Injection Piping
ISI-107, SH. 1 & 2	ISI-118, SH. 1 & 2	ISI Class 1 Reactor Water Cleanup Piping
ISI-108	ISI-119	ISI Class 1 Reactor Shutdown Cooling Piping
ISI-109	ISI-120, SH. 1 & 2	ISI Class 1 Standby Liquid Control
ISI-110	ISI-121	ISI Class 1 Control Rod Drive Hydraulic Piping
ISI-111	ISI-122	ISI Class 1 High Pressure Coolant Injection Piping
ISI-126	ISI-123	ISI Class 1 Reactor Head Vent Piping
ISI-127	ISI-124	ISI Class 1 Reactor Head Spray Piping
ISI-128, SH. 1 to 3	ISI-125, SH. 1 to 3	ISI Class 1 Reactor Pressure Vessel
ISI-200, SH. 1 & 2	ISI-205, SH. 1 & 2	ISI Class 2 Core Spray Piping
ISI-200, SH. 3	ISI-205, SH. 3 & 4	ISI Class 2 Core Spray Piping Obstructions
ISI-201	ISI-206	ISI Class 2 Isolation Condenser Piping
ISI-202, SH. 1 to 4	ISI-207, SH. 1 to 4	ISI Class 2 Low Pressure Coolant Injection Piping
ISI-202, SH. 5 & 6	ISI-207, SH. 5 & 6	ISI Class 2 Low Pressure Coolant Injection Piping Obstructions
ISI-202, SH. 7	ISI-207, SH. 7	ISI Class 2 Low Pressure Coolant Injection Piping
ISI-202, SH. 8	ISI-207, SH. 8	ISI Class 2 LPCI/CCSW Heat Exchangers
ISI-203, SH. 1 & 2	ISI-208, SH. 1 & 2	ISI Class 2 High Pressure Coolant Injection Piping
ISI-203, SH. 3 & 4	ISI-208, SH. 3 & 4	ISI Class 2 High Pressure Coolant Injection Piping Obstructions

TABLE 2.4-1
ISI ISOMETRIC AND PRESSURE TESTING DRAWINGS

UNIT 2	UNIT 3 & COMMON	TITLE
ISI-204	ISI-209	ISI Class 2 Reactor Water Cleanup and Reactor Feedwater Piping
ISI-213, SH. 1	ISI-212, SH. 1	ISI Class 2 CRD Scram Discharge Volume, East Bank
ISI-213, SH. 2	ISI-212, SH. 2	ISI Class 2 CRD Scram Discharge Volume, West Bank
ISI-214	ISI-215	ISI Class 2 ECCS Ring Header
ISI-300, SH. 1 to 6	ISI-302, SH. 1 to 4	ISI Class 3 Containment Cooling Service Water Piping
ISI-301, SH. 1 to 7	ISI-303, ISI-304, SH. 1 to 9	ISI Class 3 Diesel Generator Service Water Piping
ISI-307	ISI-305	ISI Class 3 Isolation Condenser and Vent Piping
ISI-308, SH. 1 & 2	ISI-306, SH. 1 & 2	ISI Class 3 Safety Relief Valve Discharge Piping
ISI-501, SH. 1	ISI-551, SH. 1	System Pressure Test Walkdown Isometric Reactor Head Cavity (589' Elevation)
ISI-501, SH. 2	ISI-551, SH. 2	System Pressure Test Walkdown Isometric Drywell Fourth Floor (576'-7 1/8" Elevation)
ISI-501, SH. 3	ISI-551, SH. 3	System Pressure Test Walkdown Isometric Drywell Third Floor (562'-0" Elevation)
ISI-501, SH. 4	ISI-551, SH. 4	System Pressure Test Walkdown Isometric Drywell Second Floor (537'-1 ¼" Elevation)
ISI-501, SH. 5	ISI-551, SH. 5	System Pressure Test Walkdown Isometric Drywell First Floor (515'-5 ¾" Elevation)
ISI-501, SH. 6	ISI-551, SH. 6	System Pressure Test Walkdown Isometric Drywell Basement (502'-4" Elevation)
ISI-501, SH. 7	ISI-551, SH. 7	System Pressure Test Walkdown Isometric Lower Head CRD Area (502'-4" Elevation)
ISI-501, SH. 8	ISI-551, SH. 8	System Pressure Test Walkdown Isometric Instrumentation
ISI-502, SH. 1	ISI-552, SH. 1	System Pressure Test Walkdown Isometric Isolation Condenser Piping
ISI-502, SH. 2 & 3	ISI-552, SH. 2 & 3	System Pressure Test Walkdown Isometric Isolation Condenser, Instrumentation, and Vent Piping
ISI-503	ISI-553	System Pressure Test Walkdown Isometric Shutdown Cooling Piping

TABLE 2.4-1
ISI ISOMETRIC AND PRESSURE TESTING DRAWINGS

UNIT 2	UNIT 3 & COMMON	TITLE
ISI-504	ISI-554	System Pressure Test Walkdown Isometric MSIV Room-X Area
ISI-505, SH. 1	ISI-555, SH. 1	System Pressure Test Walkdown Isometric Control Rod Drive Hydraulic Piping, East Bank
ISI-505, SH. 2	ISI-555, SH. 2	System Pressure Test Walkdown Isometric Control Rod Drive Hydraulic Piping, West Bank
ISI-506, SH. 1 to 3	ISI-556, SH. 1 to 3	System Pressure Test Walkdown Isometric Standby Liquid Control Piping
ISI-507, SH. 1 & 2	ISI-557, SH. 1 & 2	System Pressure Test Walkdown Isometric Core Spray Piping
ISI-508	ISI-558	System Pressure Test Walkdown Isometric Head Flange Seal Leak Detection
ISI-509, SH. 1 & 2	ISI-559, SH. 1 & 2	System Pressure Test Walkdown Isometric L.P. Coolant Injection Piping
ISI-509, SH. 3 & 4	ISI-559, SH. 3 & 4	System Pressure Test Walkdown Isometric Containment Cooling Service Water Piping
ISI-509, SH. 5	Not Applicable	System Pressure Test Walkdown Isometric Control Room HVAC
ISI-509, SH. 6	ISI-559, SH. 5	System Pressure Test Walkdown Isometric CCSW Supply to HPCI/LPCI Room Coolers
ISI-510, SH. 1 & 2	ISI-560, SH. 1 & 2	System Pressure Test Walkdown Isometric H. P. Coolant Injection Piping
ISI-510, SH. 3	ISI-560, SH. 3	System Pressure Test Walkdown Isometric Torus Level Instrumentation
ISI-511	ISI-561	System Pressure Test Walkdown Isometric Safety Relief Valve Discharge Piping
ISI-512	ISI-562 SH. 1 (Unit 3) & 2 (Common Unit 2/3)	System Pressure Test Walkdown Isometric Diesel Generator Service Water
ISI-513, SH. 1	ISI-563, SH. 1	System Pressure Test Walkdown Isometric Emergency Core Cooling, Torus Ring Header
ISI-513, SH. 2	ISI-563, SH. 2	System Pressure Test Walkdown Isometric ECCS Keep Fill Pump and Piping

TABLE 2.4-2
UNIT 2 CISI REFERENCE DRAWINGS

DRAWING NUMBER	TITLE
2-CISI-1000, SH. 1	IWE Component Drawing: Primary Containment General Arrangement
2-CISI-1000, SH. 2	IWE Component Rollout: Containment Shell (Drywell) View Looking Out, 0° To 360° Azimuth
2-CISI-1000, SH. 3	IWE Component Rollout: Containment Shell (Torus), 0° To 360° Azimuth
2-CISI-1000, SH. 4	IWE Component Detail: Torus Vent System - Vent Header Connection
2-CISI-1000, SH. 5	IWE Component Detail: Drywell Head

TABLE 2.4-3
UNIT 3 CISI REFERENCE DRAWINGS

DRAWING NUMBER	TITLE
3-CISI-1000, SH. 1	IWE Component Drawing: Primary Containment General Arrangement
3-CISI-1000, SH. 2A	IWE Component Rollout: Containment Shell (Drywell) View Looking Out, 0° To 360° Azimuth
3-CISI-1000, SH. 2B	IWE Component Detail: Drywell Floor Core Bore - Hole Locations And Details
3-CISI-1000, SH. 3	IWE Component Rollout: Containment Shell (Torus), 0° To 360° Azimuth
3-CISI-1000, SH. 4	IWE Component Detail: Torus Vent System - Vent Header Connection
3-CISI-1000, SH. 5	IWE Component Detail: Drywell Head

2.5 Technical Approach and Positions

When the requirements of ASME Section XI are not easily interpreted, DNPS has reviewed general licensing/regulatory requirements and industry practice to determine a practical method of implementing the Code requirements. The Technical Approach and Position (TAP) documents contained in this section have been provided to clarify DNPS's implementation of ASME Section XI requirements. An index which summarizes each TAP is included in Table 2.5-1.

TABLE 2.5-1
TECHNICAL APPROACH AND POSITIONS INDEX

Position Number	Revision Date[2]	Status[1]	(Program) Description of Technical Approach and Position
I5T-01	0 9/19/12	Active	**(SPT)** System Leakage Testing of Non-Isolable Buried Components.
I5T-02	0 9/19/12	Active	**(SPT)** Valve Seats/Discs as Pressurization Boundaries.
I5T-03	0 9/19/12	Active	**(ISI)** Reactor Coolant Makeup Systems.

Note 1: ISI Program Technical Approach and Position Status Options: Active - Current Technical Approach and Position is being utilized at DNPS; Deleted - Technical Approach and Position is no longer being utilized at DNPS.

Note 2: The revision listed is the latest revision of the subject Technical Approach and Position. The date noted in the second column is the date of the ISI Program Plan revision when the Technical Approach and Position was incorporated into the document.

TECHNICAL APPROACH AND POSITION NUMBER: I5T-01
Revision 0
(Page 1 of 1)

COMPONENT IDENTIFICATION:

Code Class:	3
Reference:	IWA-5244(b)(2)
Examination Category:	NA
Item Number:	NA
Description:	System Leakage Testing of Non-Isolable Buried Components
Component Number:	Non-Isolable Buried Pressure Retaining Components

CODE REQUIREMENT:

IWA-5244(b)(2) requires non-isolable buried components be tested to confirm that flow during operation is not impaired.

POSITION:

IWA-5000 provides no guidance in setting acceptance criteria for what can be considered "adequate flow." In lieu of any formal guidance provided by the Code, DNPS has established the following acceptance criteria:

- For open ended lines on systems that require Inservice Testing (IST) of pumps, adherence to IST acceptance criteria is considered as reasonable proof of adequate flow through the lines.

This acceptance criteria will be utilized as proof of adequate flow in order to meet the requirements of IWA-5244(b)(2).

DNPS's position is that proof of adequate flow is all that is required for testing the buried pipe segments of these open ended lines and that no further visual examination is necessary. This is consistent with the requirements for buried piping, which is not subject to visual examination.

TECHNICAL APPROACH AND POSITION NUMBER: I5T-02
Revision 0
(Page 1 of 1)

COMPONENT IDENTIFICATION:

Code Class:	1, 2, and 3
Reference:	IWA-5221, IWA-5222
Examination Category:	B-P, C-H, D-B
Item Number:	B15.10, B15.20, C7.10, D2.10
Description:	Valve Seats/Discs as Pressurization Boundaries
Component Number:	All Pressure Testing Boundary Valves

CODE REQUIREMENT:

IWA-5221 requires the pressurization boundary for system leakage testing extend to those pressure retaining components under operating pressures during normal system service.

POSITION:

DNPS's position is that the test pressurization boundary extends up to the valve seat/disc of the valve utilized for isolation. For example, in order to pressure test the Class 1 components, the valve that provides the Class break would be utilized as the isolation point. In this case the true pressurization boundary, and Class break, is actually at the valve seat/disc.

Any requirement to test beyond the valve seat/disc is dependent only on whether or not the piping on the other side of the valve seat/disc is Class 1, 2, or 3.

The extension of the pressurization boundary during a pressure test would require an abnormal valve line-up. Extending the boundary would require the over pressurization of low pressure piping at systems that have a high/low pressure interface (such as LPCI and Core Spray).

In order to simplify examination of classed components, DNPS will perform a VT-2 visual examination of the entire boundary valve body and bonnet (during pressurization up to the valve seat/disc).

TECHNICAL APPROACH AND POSITION NUMBER: I5T-03
Revision 0
(Page 1 of 2)

COMPONENT IDENTIFICATION:

Code Class:	1, 2, and 3
Reference:	IWA-4131.1(a)(2), IWB-1220(a), 10CFR50.55a(c)(2)(i)
Examination Category:	NA
Item Number:	NA
Description:	Reactor Coolant Makeup Systems
Components Number:	Class 1 Pressure Retaining Components

CODE REQUIREMENT:

IWA-4131.1(a)(2) provides size criteria for application of alternative repair replacement requirements for small items no larger than NPS 1. The alternative requirements are applied to components of a size and design such that, in the event of postulated failure during normal plant operating conditions, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by normal reactor coolant makeup systems operable from on-site emergency power.

IWB-1220(a) provides an exemption from volumetric and surface examination for components that are connected to the reactor coolant system and part of the reactor coolant pressure boundary, and that are of such a size and shape so that upon postulated rupture the resulting flow of coolant from the reactor coolant system under normal plant operating conditions is within the capacity of makeup systems that are operable from on-site emergency power. The emergency core cooling systems are excluded from the calculation of makeup capacity.

10CFR50.55a(c)(2)(i) allows components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary as defined in 10CFR50.2 need not meet the requirements for Class 1 components when, in the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system.

POSITION:

DNPS credits the control rod drive (CRD) system as the reactor coolant makeup system. DNPS UFSAR Section 3.1.2.4.4 states that during normal operation, water level in the reactor vessel is maintained by the feedwater system. For small breaks in the reactor coolant pressure boundary, the makeup capability is provided by the feedwater system and the control rod drive (CRD) system. The emergency core cooling system provides inventory makeup for breaks beyond the capability of the feedwater system.

TECHNICAL APPROACH AND POSITION NUMBER: I5T-03
Revision 0
(Page 2 of 2)

While the CRD system is not normally fed from the emergency source, it is capable of being re-aligned to emergency buses. Since the CRD system is operable from on-site emergency power, the capabilities of this system are utilized for determining size of components within the capability of the reactor coolant makeup system.

The CRD system is capable of providing makeup to the reactor coolant system utilizing CRD cooling water flow through each of the CRD units, seal purge flow through the reactor recirculation pump seals, and reactor vessel head cooling injected through the head spray nozzle in the reactor head. Since reactor vessel head cooling injected through the head spray nozzle does not supply makeup water to the reactor coolant system during normal reactor operation, this source of makeup is not credited. Therefore, the normal makeup to the reactor coolant system is as follows:

CRD Cooling Water Flow	58 gpm
Recirculation Pump Seal Purge Flow	5 gpm
	63 gpm

In determining the size of the liquid and steam lines (DNPS Calculation XCE040.0201), within the capability of the makeup system, liquid lines are defined as those which penetrate the RPV below the normal water level and steam lines as those which penetrate the RPV above the normal water level. The line diameters based on the reactor coolant system makeup capability are as follows:

Liquid Line Break Size	0.533" I.D.
Steam Line Break Size	0.876" I.D.

These line sizes are utilized when applying the following code requirements.

IWA-4131.1(a)(2) limits the application of alternative requirements for small items to those line sizes bounded by the above calculation.

IWB-1220(a) is not applied since this exemption is bound by IWB-1220(b)(1) which exempts piping NPS 1 and smaller, and IWB-1220(b)(2) which exempts components and their connections in piping of NPS 1 and smaller.

No reactor coolant pressure boundary components are excluded from Class 1 requirements in accordance with 10CFR50.55a(c)(2)(i) due to the small line sizes that can be excluded.

3.0 COMPONENT ISI PLAN

The DNPS Component ISI Plan includes ASME Section XI nonexempt pressure retaining welds, piping structural elements, pressure retaining bolting, attachment welds, pump casings, and valve bodies of ISI Class 1, 2, and 3 components that meet the criteria of Subarticle IWA-1300. These components are identified on the P&IDs listed in Section 2.3, Table 2.3-1. Procedure ER-AA-330-002, "Inservice Inspection of Section XI Welds and Components," implements the ASME Section XI Component ISI Plan. This Component ISI Plan also includes augmented inspection program requirements specified by documents other than ASME Section XI as referenced in Section 2.2.

The RPV interior, interior attachments, and welded core support structures are inspected in accordance with the Reactor Internals Program per Relief Request I5R-07.

3.1 Nonexempt ISI Class Components

The ISI Class 1, 2, and 3 nonexempt components subject to examination are those which are not exempted under the criteria in Paragraphs IWB-1220, IWC-1220, and IWD-1220, respectively. (Note: those systems which provide reactor coolant makeup and application of the exemptions are documented in Technical Approach and Position I5T-03.) A summary of DNPS Units 2, 3, and Common ASME Section XI nonexempt components is included in Section 7.0.

3.1.1 Identification of ISI Class 1, 2, and 3 Nonexempt Components

ISI Class 1, 2, and 3 nonexempt components are identified on the isometric and component drawings listed in Section 2.4, Table 2.4-1. Welded attachments are also identified by controlled DNPS individual support detail drawings.

3.2 Risk-Informed Examination Requirements

Inspections of ASME Examination Categories B-F, B-J, C-F-1 and C-F-2 components have been exempted from ASME Section XI required inspections by Relief Request I5R-02. This Relief Request allows for the implementation of a RISI Program. Piping structural elements that fall under RISI Examination Category R-A are risk ranked as High (1, 2, and 3), Medium (4 and 5), and Low (6 and 7). Per the EPRI Topical Report TR-112657, Rev. B-A and ASME Code Case N-578-1, piping structural elements ranked as High or Medium Risk are subject to examination while piping structural elements ranked as Low Risk are not subject to examinations (except for pressure testing). Thin wall welds that were excluded from volumetric examination under ASME Section XI rules per Table IWC-2500-1 are included in the element scope that is potentially subject to RISI examination at DNPS.

Piping structural elements may be excluded from examination (other than pressure testing) under the RISI Program if the only degradation mechanism present for a given location is inspected for cause under certain other DNPS programs such as the Flow Accelerated Corrosion (FAC) or Intergranular Stress Corrosion Cracking (IGSCC) Programs. These piping structural elements will remain part of the assigned programs which already perform “for cause” inspections to detect these degradation mechanisms. Piping structural elements susceptible to FAC or IGSCC along with another degradation mechanism (e.g., thermal fatigue) are retained as part of the RISI scope and are included in the element selection for the purpose of performing examinations to detect the additional degradation mechanism.

4.0 SUPPORT ISI PLAN

The DNPS Support ISI Plan includes the supports of ASME Section XI nonexempt ISI Class 1, 2, and 3 components as described in Section 3.0. Procedure ER-AA-330-003, "Inservice Inspection of Section XI Component Supports," implements the ASME Section XI Support ISI Plan.

4.1 Nonexempt ISI Class Supports

The DNPS ISI Class 1, 2, and 3 nonexempt supports are those which do not meet the exemption criteria of Paragraph IWF-1230. A summary of DNPS Units 2, 3, and Common ASME Section XI nonexempt supports is included in Section 7.0.

4.1.1 Identification of ISI Class 1, 2, and 3 Nonexempt Supports

ISI Class 1, 2, and 3 supports are identified on the ISI isometric drawings listed in Section 2.4, Table 2.4-1. Supports are also identified by controlled DNPS individual support detail drawings.

4.2 Snubber Examination and Testing Requirements

4.2.1 As allowed by 10CFR50.55a(b)(3), (b)(3)(v), and (b)(3)(v)(B), DNPS will use Subsection ISTD, "Inservice Testing of Dynamic Restraints (Snubbers) In Light Water Reactor Power Plants," ASME Operation and Maintenance of Nuclear Power Plants Code (ASME OM Code), 2004 Edition through the 2006 Addenda, to meet the visual examination, functional testing, and service life monitoring requirements for safety-related snubbers. This approach is consistent with ASME Section XI, Paragraph IWF-1220, which excludes inservice inspection of snubbers and defers to the ASME OM Code for examination, testing, and monitoring requirements. A summary of the DNPS snubber program is included in Section 7.2.

Procedure ER-AA-330-004, "Visual Examination of Snubbers," implements the visual examination of snubbers. Procedures ER-AA-330-010, "Snubber Functional Testing," and ER-AA-330-011 "Snubber Service Life Monitoring Program," implement the functional testing and service life monitoring of snubbers.

4.2.2 The ASME Section XI visual examination boundary of a support containing a snubber is defined in Figure IWF-1300-1(f). This boundary does not include the snubber pin-to-pin and does not include the connections to the snubber assembly (pins) per Paragraph IWF-1300(h).

This results in the remaining ASME Section XI requirements for VT-3 visual examination of the snubber attachment hardware including bolting and clamps. The ASME Section XI ISI Program uses Subsection IWF to define the inspection requirements for all ISI Class 1, 2, and 3 supports, regardless of type. The ISI Program maintains the Code Class snubbers in the support populations subject to inspection per Subsection IWF. This is done to facilitate scheduling, preparation including insulation removal, and inspection requirements of the snubber attachment hardware (e.g., bolting and clamps).

It should be noted that the examination of snubber welded attachments will be performed in accordance with the ASME Section XI Subsections IWB, IWC, and IWD welded attachment examination requirements (e.g.; Examination Categories B-K, C-C, and D-A).

5.0 SYSTEM PRESSURE TESTING ISI PLAN

The DNPS System Pressure Testing ISI Plan includes pressure retaining ASME Section XI, ISI Class 1, 2, and 3 components, with the exception of those specifically exempted by Paragraphs IWA-5110(c), IWC-5222(b) and IWD-5222(b). RISI piping structural elements, regardless of risk classification, remain subject to pressure testing as part of the current ASME Section XI program.

The SPT Program performs system pressure tests and required VT-2 visual examinations on the ISI Class 1, 2, and 3 pressure retaining components to verify system and component structural integrity. This program conducts both Periodic and Interval (10-year frequency) pressure tests as defined in ASME Section XI Inspection Program. Procedure ER-AA-330-001, "Section XI Pressure Testing," implements the ASME Section XI System Pressure Testing ISI Plan.

5.1 ISI Class Systems

All ISI Class 1 pressure retaining components, typically defined as the reactor coolant pressure boundary, are required to be tested. Those portions of ISI Class 2 and 3 systems that are required to be tested include the pressure retaining boundaries of components required to operate or support the system safety functions. ISI Class 2 and 3 open ended discharge piping and components are excluded from the examination requirements per Paragraphs IWC-5222(b) and IWD-5222(b).

5.1.1 Identification of ISI Class 1, 2, and 3 Components

Components subject to ASME Section XI System Pressure Testing are shown on the P&IDs listed in Section 2.3, Table 2.3-1. Additional information on the classification of various system code boundaries is provided in the ISI Classification Basis Document.

5.1.2 Identification of System Pressure Tests

System Pressure Test Walkdown Isometric Drawings are listed in Section 2.4, Table 2.4-1. Individual tests and test segments are identified and maintained in the DNPS ISI Database.

5.2 Risk-Informed Examinations of Socket Welds

Socket welds selected for examination under the RISI Program are to be inspected with a VT-2 visual examination each refueling outage per ASME Code Case N-578-1 (see footnote 12 in Table 1 of the Code Case). To facilitate this, socket welds selected for inspection under the RISI Program are pressurized each refueling outage during a system pressure test in accordance with Paragraph IWA-5211(a).

6.0 CONTAINMENT ISI PLAN

The DNPS Containment ISI Plan includes ASME Section XI ISI Class MC pressure retaining components and their integral attachments that meet the criteria of Subarticle IWA-1300. This Containment ISI Plan also includes information related to augmented examination areas, component accessibility, and examination review.

DNPS has no ISI Class CC components which meet the criteria of Subarticle IWL-1100, therefore, no requirements to perform examinations in accordance with Subsection IWL are incorporated into this Containment ISI Plan.

The inspection of containment structures and components are performed per procedures ER-AA-330-007, "Visual Examination of Section XI Class MC Surfaces and Class CC Liners," implements the Containment ISI Plan, ER-AA-335-004, "Manual Ultrasonic Measurement of Material Thickness," ER-AA-335-018, "Detailed, General, VT-1, VT-1C, VT-3, and VT-3C, Visual Examination of ASME Class MC and CC Containment Surfaces and Components."

6.1 Nonexempt ISI Class Components

The DNPS ISI Class MC components identified on the CISI Reference Drawings are those not exempted under the criteria of Paragraph IWE-1220 in the 2001 Edition through the 2003 Addenda of ASME Section XI. A summary of DNPS Units 2 and 3 ASME Section XI nonexempt CISI components is included in Section 7.0.

The process for scoping DNPS components for inclusion in the Containment ISI Plan is included in the containment sections of the ISI Classification Basis Document. These sections include a listing and detailed basis for inclusion of containment components.

Components that are classified as ISI Class MC pressure retaining components and their integral attachments must meet the requirements of ASME Section XI in accordance with 10CFR50.55a(g)(4). Class MC Supports of Subsection IWE components are not required to be examined in accordance with 10CFR50.55a(g)(4)(v). (Note that per NUREG-1796, DNPS will perform a VT-3 visual examination on nonexempt Class MC piping supports, which were added to the augmented inspection program in accordance with the DNPS commitment for license renewal.) (See Section 2.2.5 of the ISI Program Plan.)

6.1.1 Identification of ISI Class MC Nonexempt Components

ISI Class MC components are identified on the CISI Reference Drawings listed in Section 2.4, Tables 2.4-2 and 2.4-3.

6.1.2 Identification of ISI Class MC Exempt Components

Certain containment components or parts of components may be exempted from examination based on design and accessibility per the requirements of Paragraph IWE-1220.

The process for exempting DNPS components from the Containment ISI Plan per Paragraph IWE-1220 is included in the containment sections of the ISI Classification Basis Document. These sections include discussions of exempt components and the bases for those exemptions.

6.2 Augmented Examination Areas

The containment sections of the ISI Classification Basis Document discuss the containment design and components. Metal containment surface areas subject to accelerated degradation and aging require augmented examination per Examination Category E-C and Paragraph IWE-1240.

A significant condition is a condition that is identified as requiring application of additional augmented examination requirements under Paragraph IWE-1240.

In the First CISI Interval, portions of the DNPS Unit 3 Drywell Shell located at the sand pocket were identified as augmented surface areas requiring examination in accordance with Paragraph IWE-1240. These surface areas were categorized in accordance with Table IWE-2500-1, Examination Category E-C, Item Number E4.12, requiring volumetric examination of 100% of the minimum wall thickness locations identified.

6.3 Component Accessibility

ISI Class MC components subject to examination shall remain accessible for either direct or remote visual examination from at least one side per the requirements of ASME Section XI, Paragraph IWE-1230.

Paragraph IWE-1231(a)(3) requires 80% of the pressure-retaining boundary that was accessible after construction to remain accessible for either direct or remote visual examination, from at least one side of the vessel, for the life of the plant. DNPS Calculation DRE03-0032 addresses compliance with this requirement by calculating the containment pressure boundary surface area that was accessible for examination at the beginning of the CISI Program and determining the limit for surface area which may be made inaccessible for the balance of plant life.

Portions of components embedded in concrete or otherwise made inaccessible during construction are exempted from examination, provided that the requirements of ASME Section XI, Paragraph IWE-1232 have been fully satisfied.

In addition, inaccessible surface areas exempted from examination include those surface areas where visual access by line of sight with adequate lighting from permanent vantage points is obstructed by permanent plant structures, equipment, or components; provided these surface areas do not require examination in accordance with the inspection plan, or augmented examination in accordance with Paragraph IWE-1240.

6.4 Responsible Individual

ASME Section XI Subsection IWE requires the Responsible Individual to be involved in the development, performance, and review of the CISI examinations. The Responsible Individual shall meet the requirements of ASME Section XI, Paragraph IWE-2320.

7.0 COMPONENT SUMMARY TABLES

7.1 Inservice Inspection Summary Tables

The following Tables 7.1-1 and 7.1-2 provide a summary of the ASME Section XI pressure retaining components, supports, containment structures, system pressure testing, and augmented inspection program components for the Fifth ISI Interval and the Second CISI Interval at DNPS Units 2, 3, and Common.

The format of the Inservice Inspection Summary Tables is as depicted below and provides the following information:

Examination Category (with Examination Category Description)	Item Number (or Risk Category Number or Augmented Number)	Description	Exam Requirements	Total Number of Components by System (or Test Block Number)	Approved Relief Request/ TAP Number	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)

(1) Examination Category (with Examination Category Description):

Provides the examination category and description as identified in ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, IWE-2500-1, and IWF-2500-1. Only those examination categories applicable to DNPS are identified.

Examination Category “R-A” from ASME Code Case N-578-1 is used in lieu of ASME Section XI Examination Categories B-F, B-J, C-F-1, and C-F-2 to identify ISI Class 1 and 2 piping structural elements for the RISI Program.

Examination Category “NA” is used to identify augmented inspection programs and other DNPS requirements.

(2) Item Number (or Risk Category Number or Augmented Number):

Provides the item number as identified in ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, IWE-2500-1, and IWF-2500-1. Only those item numbers applicable to DNPS are identified.

For piping structural elements under the RISI Program, the Risk Category Number (e.g., 1 through 5) is used in place of the Item Number.

Specific abbreviations such as BWROG, FSSE, IGSCC, 1796.27, and 1796.30 have been developed to identify augmented inspection programs and other DNPS requirements.

(3) Item Number (or Risk Category or Augmented) Description:

Provides the description as identified in ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, IWE-2500-1, and IWF-2500-1.

For Risk-Informed piping structural elements, a statement of the Risk Category is provided.

For augmented inspection programs, a description of the augmented basis is provided.

(4) Examination Requirements:

Provides the examination methods required by ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, IWE-2500-1, and IWF-2500-1.

Provides the examination requirements for piping structural elements under the RISI Program that are in accordance with the EPRI Topical Report TR-112657, Rev. B-A and ASME Code Case N-578-1.

Provides the examination requirements for augmented inspection program components.

(5) Total Number Of Components by System (or Test Block Number):

Provides the system designator (abbreviations). See Section 2.3, Table 2.3-1 for a list of these systems.

This column also provides the number of components within a particular system for that Item Number, Risk Category Number, or Augmented Number.

Provides the unique alphanumeric identification number of each test block for System Pressure Testing Categories. The number consists of the unit designation (2, 3, or 2/3), a two letter system designation, followed by two numeric characters that uniquely identify the test block within the system (i.e., 2CS01 would be the number 01 test block in the Unit 2 Core Spray system).

Note that the total number of components by system are subject to change after plant modifications, design changes, and ISI system classification updates.

(6) Approved Relief Request/TAP Number:

Provides a listing of Approved Relief Request/TAP Numbers applicable to specific components, the ASME Section XI Item Number, Risk Category Number, or Augmented Number. Relief Requests and TAP Numbers that generically apply to all components, or an entire class are not listed. If a Relief

Request/TAP Number is identified, see the corresponding relief request in Section 8.0 or the TAP Number in Section 2.5.

(7) Notes:

Provides a listing of program notes applicable to the ASME Section XI Item Number, Risk Category Number, or Augmented Number. If a program note number is identified, see the corresponding program note at the end of the Table 7.1-3.

TABLE 7.1-1 - UNIT 2
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
B-A Pressure Retaining Welds in Reactor Vessel	B1.11	Circumferential Shell Welds (Reactor Vessel)	Volumetric	RPV: 4	I5R-06	
	B1.12	Longitudinal Shell Welds (Reactor Vessel)	Volumetric	RPV: 18	I5R-06	
	B1.21	Circumferential Head Welds (Reactor Vessel)	Volumetric	RPV: 1		
	B1.22	Meridional Head Welds (Reactor Vessel)	Volumetric	RPV: 6		
	B1.30	Shell-to-Flange Weld (Reactor Vessel)	Volumetric	RPV: 1		
	B1.40	Head-to-Flange Weld (Reactor Vessel)	Volumetric & Surface	RPV: 1		
B-D Full Penetration Welds of Nozzles in Vessels	B3.90	Nozzle-to-Vessel Welds (Reactor Vessel)	Volumetric	RPV: 31		11
	B3.100	Nozzle Inside Radius Section (Reactor Vessel)	Volumetric	RPV: 31	I5R-01	
B-G-1 Pressure Retaining Bolting, Greater Than 2 in. In Diameter	B6.10	Closure Head Nuts (Reactor Vessel)	Visual, VT-1	RPV: 1 (92 Nuts)		
	B6.20	Closure Studs (Reactor Vessel)	Volumetric	RPV: 1 (92 Studs)		6
	B6.40	Threads in Flange (Reactor Vessel)	Volumetric	RPV:1 (92 Threads)		
	B6.50	Closure Washers, Bushings (Reactor Vessel)	Visual, VT-1	RPV: 1 (92 Washers, Bushings)		
	B6.180	Bolts and Studs (Pumps)	Volumetric	RR: 2		
	B6.190	Flange Surface, when connection disassembled (Pumps)	Visual, VT-1	RR: 2		
	B6.200	Nuts, Bushings, and Washers (Pumps)	Visual, VT-1	RR: 2		

TABLE 7.1-1 - UNIT 2
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
B-G-2 Pressure Retaining Bolting, 2 in. and Less In Diameter	B7.50	Bolts, Studs, and Nuts (Piping)	Visual, VT-1	ISCO: 1 MS: 13 RHS: 4 RHSP: 1 RHV: 3 RR: 4 RVBD: 1 SBLC: 2 SDC: 5		
	B7.70	Bolts, Studs, and Nuts (Valves)	Visual, VT-1	LPCI: 2 MS: 8 RR: 6		

TABLE 7.1-1 - UNIT 2
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
B-K Welded Attachments for Vessels, Piping, Pumps, and Valves	B10.10	Welded Attachments (Pressure Vessels)	Surface or Volumetric	RPV: 5		
	B10.20	Welded Attachments (Piping)	Surface	CS: 7 FW: 10 HPCI: 5 ISCO: 5 LPCI: 6 MS: 23 RHS: 2 RHV: 4 RR: 9 RWCU: 3 SBLC: 2 SDC: 10		
	B10.30	Welded Attachments (Pumps)	Surface	RR: 6		
	B10.40	Welded Attachments (Valves)	Surface	RR: 2		
B-L-2 Pump Casings	B12.20	Pump Casing (Pumps)	Visual, VT-3	RR: 2		
B-M-2 Valve Bodies	B12.50	Valve Body (Exceeding NPS 4) (Valves)	Visual, VT-3	CS: 6 FW: 6 HPCI: 2 ISCO: 4 LPCI: 6 MS: 21 RR: 6 RWCU: 3 SDC: 7		

TABLE 7.1-1 - UNIT 2
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
B-N-1 Interior of Reactor Vessel	B13.10	Vessel Interior (Reactor Vessel)	Visual, VT-3	RPV: 1	I5R-07	12
B-N-2 Welded Core Support Structures and Interior Attachments to Reactor Vessels	B13.20	Interior Attachments Within Beltline Region (Reactor Vessel)	Visual, VT-1	RPV: 26	I5R-07	12
	B13.30	Interior Attachments Beyond Beltline Region (Reactor Vessel)	Visual, VT-3	RPV: 33	I5R-07	12
	B13.40	Core Support Structure (Reactor Vessel)	Visual, VT-3	RPV: 1	I5R-07	12
B-O Pressure Retaining Welds in Control Rod Housings	B14.10	Welds in CRD Housing (Reactor Vessel) (10% of Peripheral CRD Housings to be inspected. 32 of the 177 CRD Housings are identified as peripheral)	Volumetric or Surface	RPV: 32 (32 CRD Housings with 2 Welds Each)		10

TABLE 7.1-1 - UNIT 2
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Test Block Number	Approved Relief Request/ TAP Number	Notes
B-P All Pressure Retaining Components (Periodic)	B15.10	Pressure Retaining Components [IWB-5222(a)] System Leakage Test (IWB-5220)	Visual, VT-2	2RC01 2SC01	I5T-02	
B-P All Pressure Retaining Components (Interval)	B15.20	Pressure Retaining Components [IWB-5222(b)] System Leakage Test (IWB-5220)	Visual, VT-2	2RC01 2SC01	I5T-02	

TABLE 7.1-1 - UNIT 2
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
C-A Pressure Retaining Welds in Pressure Vessels	C1.30	Tubesheet-to-Shell Welds (Pressure Vessels)	Volumetric	LPCI: 4		
C-B Pressure Retaining Nozzle Welds in Vessels	C2.21	Nozzle-to-Shell (or Head or Nozzle) Welds without Reinforcing Plates in Vessels, Greater than 1/2" Nominal Thickness (Pressure Vessels)	Volumetric & Surface	ISCO: 4		
	C2.31	Reinforcing Plate Welds to Nozzle & Vessel for Nozzles with Reinforcing Plates in Vessels, Greater than 1/2" Nominal Thickness (Pressure Vessels)	Surface	ECCS: 4 LPCI: 8		
	C2.33	Nozzle-to-Shell (or Head or Nozzle) Welds with Reinforcing Plates when Inside of Vessel is Inaccessible for Vessels, Greater than 1/2" Nominal Thickness (Pressure Vessels)	Visual, VT-2	ECCS: 4 LPCI: 4		
C-C Welded Attachments For Vessels, Piping, Pumps, and Valves	C3.10	Welded Attachments (Pressure Vessels)	Surface	LPCI: 4		
	C3.20	Welded Attachments (Piping)	Surface	CRD: 2 CS: 14 ECCS: 16 HPCI: 17 ISCO: 2 LPCI: 25 RWCU: 1		
	C3.30	Welded Attachments (Pumps)	Surface	CS: 2 HPCI: 1 LPCI: 4		

TABLE 7.1-1 - UNIT 2
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Test Block Number	Approved Relief Request/ TAP Number	Notes
C-H All Pressure Retaining Components (Periodic)	C7.10	Pressure Retaining Components System Leakage Test (IWC-5220)	Visual, VT-2	2CS01 2EC01 2EC02 2EC03 2HP01 2HP02 2HP03 2HP04 2HP05 2LP01 2LP02 2LP03 2LP04 2NB01 2RC01 2RC02 2SC01 2SC02 2SC03	I5R-03 I5R-05 I5T-02	

TABLE 7.1-1 - UNIT 2
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
D-A Welded Attachments For Vessels, Piping, Pumps, and Valves	D1.10	Welded Attachments (Pressure Vessels)	Visual, VT-1	ISCO: 3		
	D1.20	Welded Attachments (Piping)	Visual, VT-1	CCSW: 3 SRVD: 16		

TABLE 7.1-1 - UNIT 2
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Test Block Number	Approved Relief Request/ TAP Number	Notes
D-B All Pressure Retaining Components (Periodic)	D2.10	Pressure Retaining Components System Leakage Test (IWD-5220)	Visual, VT-2	2CC01 2CC02 2DG01 2DG03 2IC01 2IC02	I5R-04 I5T-01 I5T-02	

TABLE 7.1-1 - UNIT 2
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components	Approved Relief Request/ TAP Number	Notes
E-A Containment Surfaces	E1.11	Containment Vessel Pressure Retaining Boundary - Accessible Surface Areas	General Visual	309		
	E1.11	Containment Vessel Pressure Retaining Boundary - Bolted Connections, Surfaces	Visual, VT-3	41		7
	E1.12	Containment Vessel Pressure Retaining Boundary - Wetted Surfaces of Submerged Areas	Visual, VT-3	16		8
	E1.20	Containment Vessel Pressure Retaining Boundary - BWR Vent System Accessible Surface Areas	Visual, VT-3	45		8
	E1.30	Moisture Barriers	General Visual	4		
E-C Containment Surfaces Requiring Augmented Examination	E4.11	Containment Surface Areas - Visible Surfaces	Visual, VT-1	0		9
	E4.12	Containment Surface Areas - Surface Area Grid Minimum Wall Thickness Location	Volumetric (UT Thickness)	0		

TABLE 7.1-1 - UNIT 2
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
F-A Supports	F1.10	Class 1 Piping Supports	Visual, VT-3	CS: 15 FW: 14 HPCI: 8 ISCO: 7 LPCI: 8 MS: 44 RHS: 8 RHV: 11 RR: 21 RVBD: 4 RWCU: 8 SBLC: 13 SDC: 16		1
	F1.20	Class 2 Piping Supports	Visual, VT-3	CRD: 28 CS: 61 ECCS: 34 HPCI: 86 ISCO: 21 LPCI: 81 RWCU: 1		1
	F1.30	Class 3 Piping Supports	Visual, VT-3	CCSW: 116 DGSW: 71 ISCO: 3 SRVD: 47		1

TABLE 7.1-1 - UNIT 2
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
F-A Supports (Continued)	F1.40	Supports Other Than Piping Supports (Class 1, 2, 3, and MC)	Visual, VT-3	CCSW: 4 CS: 2 DGSW: 1 HPCI: 2 LPCI: 8 RPV: 5 RR: 22		1

TABLE 7.1-1 - UNIT 2
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Risk Category Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
R-A Risk-Informed Piping Examinations	1	Risk Category 1 Elements	See Notes	FW: 18 MS: 3	I5R-02	3 5
	2	Risk Category 2 Elements	See Notes	CS: 19 ISCO: 11 LPCI: 34 RVBD: 7 SBLC: 13 SDC: 47	I5R-02	3 5
	3	Risk Category 3 Elements	See Notes	FW: 7 MS: 54	I5R-02	3 5
	4	Risk Category 4 Elements	See Notes	CS: 37 ECCS: 56 HPCI: 12 ISCO: 4 LPCI: 55 MS: 117 RPV: 4 RR: 69 RVBD: 22 RWCU: 15 SBLC: 35	I5R-02	3 5
	5	Risk Category 5 Elements	See Notes	HPCI: 32 RPV: 6	I5R-02	3 5

TABLE 7.1-1 - UNIT 2
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
NA Augmented Components	BWROG	BWROG, Alternate BWR Feedwater Nozzle Inspection Requirement Components	Volumetric	RPV: 8		
	FSSE	FSSE Augmented Inspection Program	Volumetric	RPV: 2 RR: 12		
	IGSCC	Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping Components, TR-113932, “BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)”, and TR-1012621, “BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75-A)”	Volumetric	Category C: 94 Category D: 64 Category E: 38 Category F: 3 Category G: 2		4
	1796.27	Class MC Component Supports (NUREG-1796, Section B.1.27)	Visual, VT-3	BS: 1 DW: 3 SC: 6 TV: 1		13
	1796.30	Nonexempt Class MC Piping Supports (NUREG-1796, Section B.1.30)	Visual, VT-3	PS: 26 RBCCW: 6		14
	1801	Cast Austenitic Stainless Steel Components (NUREG-1801)	Enhanced Visual, EVT-1	RPV: 3		15

TABLE 7.1-2 - UNIT 3 & COMMON
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
B-A Pressure Retaining Welds in Reactor Vessel	B1.11	Circumferential Shell Welds (Reactor Vessel)	Volumetric	RPV: 4	I5R-06	
	B1.12	Longitudinal Shell Welds (Reactor Vessel)	Volumetric	RPV: 14	I5R-06	
	B1.21	Circumferential Head Welds (Reactor Vessel)	Volumetric	RPV: 1		
	B1.22	Meridional Head Welds (Reactor Vessel)	Volumetric	RPV: 6		
	B1.30	Shell-to-Flange Weld (Reactor Vessel)	Volumetric	RPV: 1		
	B1.40	Head-to-Flange Weld (Reactor Vessel)	Volumetric & Surface	RPV: 1		
	B1.51	Beltline Region Repair Welds (Reactor Vessel)	Volumetric	RPV: 2		
B-D Full Penetration Welds of Nozzles in Vessels	B3.90	Nozzle-to-Vessel Welds (Reactor Vessel)	Volumetric	RPV: 31		11
	B3.100	Nozzle Inside Radius Section (Reactor Vessel)	Volumetric	RPV: 31	I5R-01	
B-G-1 Pressure Retaining Bolting, Greater Than 2 in. In Diameter	B6.10	Closure Head Nuts (Reactor Vessel)	Visual, VT-1	RPV: 1 (92 Nuts)		
	B6.20	Closure Studs (Reactor Vessel)	Volumetric	RPV: 1 (92 Studs)		6
	B6.40	Threads in Flange (Reactor Vessel)	Volumetric	RPV: 1 (92 Threads)		
	B6.50	Closure Washers, Bushings (Reactor Vessel)	Visual, VT-1	RPV: 1 (92 Washers, Bushings)		
	B6.180	Bolts and Studs (Pumps)	Volumetric	RR: 2		
	B6.190	Flange Surface, when connection disassembled (Pumps)	Visual, VT-1	RR: 2		
	B6.200	Nuts, Bushings, and Washers (Pumps)	Visual, VT-1	RR: 2		

TABLE 7.1-2 - UNIT 3 & COMMON
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
B-G-2 Pressure Retaining Bolting, 2 in. and Less In Diameter	B7.50	Bolts, Studs, and Nuts (Piping)	Visual, VT-1	ISCO: 1 MS: 13 RHS: 4 RHSP: 1 RHV: 3 RR: 6 RWCU: 2 SBLC: 2 SDC: 5		
	B7.60	Bolts, Studs, and Nuts (Pumps)	Visual, VT-1	RR: 1		
	B7.70	Bolts, Studs, and Nuts (Valves)	Visual, VT-1	LPCI: 2 MS: 8 RR: 4		

TABLE 7.1-2 - UNIT 3 & COMMON
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
B-K Welded Attachments for Vessels, Piping, Pumps, and Valves	B10.10	Welded Attachments (Pressure Vessels)	Surface or Volumetric	RPV: 5		
	B10.20	Welded Attachments (Piping)	Surface	CS: 3 FW: 14 HPCI: 7 ISCO: 4 LPCI: 2 MS: 23 RHS: 2 RR: 4 RWCU: 3 SDC: 4		
	B10.30	Welded Attachments (Pumps)	Surface	RR: 6		
	B10.40	Welded Attachments (Valves)	Surface	RR: 2		
B-L-2 Pump Casings	B12.20	Pump Casing (Pumps)	Visual, VT-3	RR: 2		
B-M-2 Valve Bodies	B12.50	Valve Body (Exceeding NPS 4) (Valves)	Visual, VT-3	CS: 6 FW: 6 HPCI: 2 ISCO: 4 LPCI: 6 MS: 21 RR: 4 RWCU: 4 SDC: 7		

TABLE 7.1-2 - UNIT 3 & COMMON
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
B-N-1 Interior of Reactor Vessel	B13.10	Vessel Interior (Reactor Vessel)	Visual, VT-3	RPV: 1	I5R-07	12
B-N-2 Welded Core Support Structures and Interior Attachments to Reactor Vessels	B13.20	Interior Attachments Within Beltline Region (Reactor Vessel)	Visual, VT-1	RPV: 46	I5R-07	12
	B13.30	Interior Attachments Beyond Beltline Region (Reactor Vessel)	Visual, VT-3	RPV: 33	I5R-07	12
	B13.40	Core Support Structure (Reactor Vessel)	Visual, VT-3	RPV: 1	I5R-07	12
B-O Pressure Retaining Welds in Control Rod Housings	B14.10	Welds in CRD Housing (Reactor Vessel) (10% of Peripheral CRD Housings to be inspected. 32 of the 177 CRD Housings are identified as peripheral)	Volumetric or Surface	RPV: 32 (32 CRD Housings with 2 Welds Each)		10

TABLE 7.1-2 - UNIT 3 & COMMON
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Test Block Number	Approved Relief Request/ TAP Number	Notes
B-P All Pressure Retaining Components (Periodic)	B15.10	Pressure Retaining Components [IWB-5222(a)] System Leakage Test (IWB-5220)	Visual, VT-2	3RC01 3SC01	I5T-02	
B-P All Pressure Retaining Components (Interval)	B15.20	Pressure Retaining Components [IWB-5222(b)] System Leakage Test (IWB-5220)	Visual, VT-2	3RC01 3SC01	I5T-02	

TABLE 7.1-2 - UNIT 3 & COMMON
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
C-A Pressure Retaining Welds in Pressure Vessels	C1.30	Tubesheet-to-Shell Weld (Pressure Vessels)	Volumetric	LPCI: 4		
C-B Pressure Retaining Nozzle Welds in Vessels	C2.21	Nozzle-to-Shell (or Head or Nozzle) Weld without Reinforcing Plates in Vessels, Greater than 1/2" Nominal Thickness (Pressure Vessels)	Volumetric & Surface	ISCO: 4		
	C2.31	Reinforcing Plate Welds to Nozzle and Vessel for Nozzles with Reinforcing Plates in Vessels, Greater than 1/2" Nominal Thickness (Pressure Vessels)	Surface	ECCS: 4 LPCI: 8		
	C2.33	Nozzle-to-Shell (or Head or Nozzle) Welds with Reinforcing Plates when Inside of Vessel is Inaccessible for Vessels, Greater than 1/2" Nominal Thickness (Pressure Vessels)	Visual, VT-2	ECCS: 4 LPCI: 4		
C-C Welded Attachments For Vessels, Piping, Pumps, and Valves	C3.10	Welded Attachments (Pressure Vessels)	Surface	LPCI: 4		
	C3.20	Welded Attachments (Piping)	Surface	CRD: 7 CS: 16 ECCS: 16 FW: 1 HPCI: 17 ISCO: 2 LPCI: 25		
	C3.30	Welded Attachments (Pumps)	Surface	CS: 2 HPCI: 1 LPCI: 4		

TABLE 7.1-2 - UNIT 3 & COMMON
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Test Block Number	Approved Relief Request/ TAP Number	Notes
C-H All Pressure Retaining Components (Periodic)	C7.10	Pressure Retaining Components System Leakage Test (IWC-5220)	Visual, VT-2	3CS01 3EC01 3EC02 3EC03 3HP01 3HP02 3HP03 3HP04 3LP01 3LP02 3LP03 3LP04 3NB01 3RC01 3RC02 3SC01 3SC02 3SC03	I5R-03 I5R-05 I5T-02	

TABLE 7.1-2 - UNIT 3 & COMMON
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
D-A Welded Attachments For Vessels, Piping, Pumps, and Valves	D1.10	Welded Attachments (Pressure Vessels)	Visual, VT-1	ISCO: 3		
	D1.20	Welded Attachments (Piping)	Visual, VT-1	CCSW: 13 DGSW: 0+1 SRVD: 16		2

TABLE 7.1-2 - UNIT 3 & COMMON
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Test Block Number	Approved Relief Request/ TAP Number	Notes
D-B All Pressure Retaining Components (Periodic)	D2.10	Pressure Retaining Components System Leakage Test (IWD-5220)	Visual, VT-2	3CC01 3DG01 3DG03 2/3DG01 2/3DG03 3IC01 3IC02	I5R-04 I5T-01 I5T-02	

TABLE 7.1-2 - UNIT 3 & COMMON
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components	Approved Relief Request/ TAP Number	Notes
E-A Containment Surfaces	E1.11	Containment Vessel Pressure Retaining Boundary - Accessible Surface Areas	General Visual	241		
	E1.11	Containment Vessel Pressure Retaining Boundary - Bolted Connections, Surfaces	Visual, VT-3	36		7
	E1.12	Containment Vessel Pressure Retaining Boundary - Wetted Surfaces of Submerged Areas	Visual, VT-3	16		8
	E1.20	Containment Vessel Pressure Retaining Boundary - BWR Vent System Accessible Surface Areas	Visual, VT-3	45		8
	E1.30	Moisture Barriers	General Visual	4		
E-C Containment Surfaces Requiring Augmented Examination	E4.11	Containment Surface Areas - Visible Surfaces	Visual, VT-1	0		9
	E4.12	Containment Surface Areas - Surface Area Grid Minimum Wall Thickness Location	Volumetric (UT Thickness)	22		

TABLE 7.1-2 - UNIT 3 & COMMON
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
F-A Supports	F1.10	Class 1 Piping Supports	Visual, VT-3	CS: 12 FW: 18 HPCI: 10 ISCO: 7 LPCI: 8 MS: 43 RHS: 6 RHV: 4 RR: 13 RVBD: 3 RWCU: 11 SBLC: 33 SDC: 13		1
	F1.20	Class 2 Piping Supports	Visual, VT-3	CRD: 29 CS: 53 ECCS: 34 FW: 1 HPCI: 69 ISCO: 23 LPCI: 80 RWCU: 1		1
	F1.30	Class 3 Piping Supports	Visual, VT-3	CCSW: 108 DGSW: 78+80 ISCO: 3 SRVD: 47		1 2

**TABLE 7.1-2 - UNIT 3 & COMMON
INSERVICE INSPECTION SUMMARY TABLE**

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
F-A Supports (Continued)	F1.40	Supports Other Than Piping Supports (Class 1, 2, 3, and MC)	Visual, VT-3	CCSW: 4 CS: 2 DGSW: 1+1 HPCI: 2 LPCI: 12 RPV: 5 RR: 22		1 2

TABLE 7.1-2 - UNIT 3 & COMMON
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Risk Category Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
R-A Risk-Informed Piping Examinations	1	Risk Category 1 Elements	See Notes	FW: 18 MS: 5	I5R-02	3 5
	2	Risk Category 2 Elements	See Notes	CS: 9 ISCO: 4 LPCI: 16 SBLC: 17 SDC: 44	I5R-02	3 5
	3	Risk Category 3 Elements	See Notes	FW: 7 MS: 62	I5R-02	3 5
	4	Risk Category 4 Elements	See Notes	CS: 51 ECCS: 56 HPCI: 9 ISCO: 3 LPCI: 40 MS: 109 RPV: 4 RR: 94 RVBD: 27 RWCU: 15 SBLC: 57	I5R-02	3 5
	5	Risk Category 5 Elements	See Notes	HPCI: 29 RPV: 6	I5R-02	3 5

TABLE 7.1-2 - UNIT 3 & COMMON
INSERVICE INSPECTION SUMMARY TABLE

Examination Category (with Examination Category Description)	Item Number	Description	Exam Requirements	Total Number of Components by System	Approved Relief Request/ TAP Number	Notes
NA Augmented Components	BWROG	BWROG, Alternate BWR Feedwater Nozzle Inspection Requirement Components	Volumetric	RPV: 8		
	IGSCC	Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping Components, TR-113932, "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)", and TR-1012621, "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75-A)"	Volumetric	Category C: 50 Category D: 13		4
	1796.27	Class MC Component Supports (NUREG-1796, Section B.1.27)	Visual, VT-3	BS: 1 DW: 3 SC: 6 TV: 1		13
	1796.30	Nonexempt Class MC Piping Supports (NUREG-1796, Section B.1.30)	Visual, VT-3	PS: 20 RBCCW: 6		14
	1801	Cast Austenitic Stainless Steel Components (NUREG-1801)	Enhanced Visual, EVT-1	RPV:3		15

TABLE 7.1-3
INSERVICE INSPECTION SUMMARY TABLE PROGRAM NOTES

Note #	Note Summary
1	Snubber visual examinations, functional testing, and service life monitoring are performed in accordance with the ASME OM Code, Subsection ISTD. For a detailed discussion of the DNPS Snubber Program, refer to Section 4.2 of this document.
2	The Unit 3 population counts include those components that are common to both units (typically designated as “2/3”) and are listed in the Table following a “+” symbol.
3	For the Fifth ISI Interval, DNPS’s ISI Class 1 and 2 piping inspection program will be governed by risk-informed regulations. The RISI Program methodology is described in the EPRI Topical Report TR-112657, Rev. B-A and ASME Code Case N-578-1. The RISI Program scope has been implemented as an alternative to the 2007 Edition through the 2008 Addenda of the ASME Section XI Code examination program for ISI Class 1 B-F and B-J welds and ISI Class 2 C-F-1 and C-F-2 welds in accordance with 10CFR50.55a(a)(3)(i).
4	Per the EPRI Topical Report TR-112657, Rev. B-A and ASME Code Case N-578-1, welds within the plant that are assigned to IGSCC Categories B through G will continue to meet existing IGSCC schedules, while IGSCC Category A welds have been subsumed into the RISI Program.
5	Examination requirements within the RISI Program are determined by the various degradation mechanisms present at each individual piping structural element. See EPRI TR-112657, Rev. B-A and ASME Code Case N-578-1 for specific examination method requirements.
6	Examination Category B-G-1, Item Numbers B6.20 “Closure Studs, In Place” and B6.30 “Closure Studs, When Removed” have been combined into and renamed as Item Number B6.20 “Closure Studs”.
7	Bolted connections examined per Item Number E1.11 require a General Visual examination each period and a VT-3 visual examination once per interval and each time the connection is disassembled during a scheduled Item Number E1.11 examination. Additionally, a VT-1 visual examination shall be performed if degradation or flaws are identified during the VT-3 visual examination. These modifications are required by 10CFR50.55a(b)(2)(ix)(G) and 10CFR50.55a(b)(2)(ix)(H).
8	Item Numbers E1.12 and E1.20 require VT-3 visual examination in lieu of General Visual examination, as modified by 10CFR50.55a(b)(2)(ix)(G).
9	Item Number E4.11 requires VT-1 visual examination in lieu of Detailed Visual examination, as modified by 10CFR50.55a(b)(2)(ix)(G).
10	Examination Category B-O (Pressure-Retaining Welds In Control Rod Housings), Item Number B14.10 (Welds in CRD Housing) - the scope of examination is for pressure retaining welds in 10% of the peripheral CRD Housings. A total of 32 out of the 177 CRD Housings are classified as peripheral components. DNPS has selected the welds on 4 CRD Housings (two welds per housing) to be examined during the interval (10% of 32).
11	As allowed by ASME Code Case N-613-1, DNPS will perform a volumetric examination using a reduced examination volume (A-B-C-D-E-F-G-H) of Figures 1, 2, and 3 of the Code Case in lieu of the previous examination volumes of ASME Section XI, Figures IWB-2500-7(a), (b), and (c).
12	The RPV interior requires examination per the BWRVIP in lieu of ASME Section XI Examination Categories B-N-1 and B-N-2 per Relief Request I5R-07. Augmented inspection programs associated with the BWRVIP and the DNPS Vessel Internals Program are maintained independently. The DNPS BWRVIP procedure ER-AB-331, “BWR Internals Program Management” describes these requirements and the BWRVIP in more detail.
13	Per NUREG-1796, DNPS will perform a VT-3 visual examination on Class MC component supports. These component supports were added to the augmented inspection program in accordance with the DNPS commitment for license renewal. (NUREG-1796, Section B.1.27)

TABLE 7.1-3
INSERVICE INSPECTION SUMMARY TABLE PROGRAM NOTES

Note #	Note Summary
14	Per NUREG-1796, DNPS will perform a VT-3 visual examination on nonexempt Class MC piping supports. These piping supports were added to the augmented inspection program in accordance with the DNPS commitment for license renewal. (NUREG-1796, Section B.1.30)
15	Per NUREG-1801, DNPS will perform a EVT-1 visual examination on a sample of the identified CASS components every outage when accessible due to Control Rod Blade (CRB) exchange. If there is no CRB exchange, than at least one fuel support piece will be inspected every other refueling outage. This requirement was added in response to the License Renewal Requirement and implemented starting in Fall 2011. (Ref. NUREG-1801 and EC 376712.)

7.2 Snubber Inspection Summary Tables

10CFR50.55a "Codes and Standards" allows usage of ASME OM Code, Subsection ISTD, using VT-3 visual examination methods described in Paragraph IWA-2213.

The following Tables 7.2-1 and 7.2-2 provide a summary of the ASME OM Code, Subsection ISTD, Snubber examinations and testing for the Fifth ISI Interval at DNPS Units 2, 3 and Common.

The format of the Snubber Inspection Summary Tables is as depicted below and provides the following information:

ASME OM Code Subsection (with Subsection Description)	OM Article Number	Article Number Description	Exam Requirements	Totals	Frequency	Approved Relief Request/ TAP Number	Notes
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)

(1) ASME OM Code Subsection:

Provides the applicable ASME OM Code subsection number and a description as obtained from Subsection ISTD. Only applicable subsections to DNPS are identified.

(2) OM Article Number:

Provides the article number as identified in Subsection ISTD. Only those article numbers applicable to DNPS are identified.

(3) Article Number Description:

Provides the article description as identified in Subsection ISTD. Identifies the methods selected to be performed at DNPS.

(4) Examination Requirements:

Provides the examination and test method(s) required by Subsection ISTD.

(5) Totals:

Provides the total number of snubbers that pertain to that article of Subsection ISTD. Note that the total number of snubbers are subject to change after completion of plant modifications and design changes.

(6) Frequency:

Provides the frequency for examinations and testing as addressed in Subsection ISTD and approved ASME OM Code Cases.

(7) Approved Relief Request/TAP Number:

Provides a listing of Approved Relief Request/TAP Numbers to specific snubber components. Relief requests and TAP Numbers that generically apply to all components, or an entire class are not listed. If a Relief Request/TAP Number is identified, see the corresponding relief request in Section 8.0 or the TAP Number in Section 2.5.

(8) Notes:

Provides a listing of program notes applicable to the Subsection ISTD article number. If a program note number is identified, see the corresponding program note at the end of the Table 7.2-3.

TABLE 7.2-1 - UNIT 2
SNUBBER INSPECTION SUMMARY TABLE

ASME OM Code Subsection (with Subsection Description)	OM Article Number	Article Number Description	Exam Requirements	Totals	Frequency	Approved Relief Request/ TAP Number	Notes
ISTD Snubber Examinations	ISTD-4200	Accessible and Inaccessible Snubbers (1 population)	Visual, VT-3	92	Various		1 3
ISTD Snubber Testing	ISTD-5200	10% Functional Test Plan - Type 1 Snubbers (PSA-1/4, PSA-1/2)	Functional Testing	3	Every Outage		2
		10% Functional Test Plan - Type 2 Snubbers (PSA-1, PSA-3, PSA-10)	Functional Testing	35	Every Outage		2
		10% Functional Test Plan - Type 3 Snubbers (PSA-35, PSA-100)	Functional Testing	40	Every Outage		2
		10% Functional Test Plan - Type 4 Snubbers (LISEGA 30 Series-1, 4, 5, & 6)	Functional Testing	14	Every Outage		2

TABLE 7.2-2 - UNIT 3 & COMMON
SNUBBER INSPECTION SUMMARY TABLE

ASME OM Code Subsection (with Subsection Description)	OM Article Number	Article Number Description	Exam Requirements	Totals	Frequency	Approved Relief Request/ TAP Number	Notes
ISTD Snubber Examinations	ISTD-4200	Accessible and Inaccessible Snubbers (1 population)	Visual, VT-3	101	Various		1 3
ISTD Snubber Testing	ISTD-5200	10% Functional Test Plan - Type 1 Snubbers (PSA-1/4, PSA-1/2)	Functional Testing	8	Every Outage		2
		10% Functional Test Plan - Type 2 Snubbers (PSA-1, PSA-3, PSA-10)	Functional Testing	44	Every Outage		2
		10% Functional Test Plan - Type 3 Snubbers (PSA-35, PSA-100)	Functional Testing	31	Every Outage		2
		10% Functional Test Plan - Type 4 Snubbers (LISEGA 30 Series-1, 4, 5, & 6)	Functional Testing	18	Every Outage		2

TABLE 7.2-3
SNUBBER INSPECTION SUMMARY TABLE PROGRAM NOTES

Note #	Note Summary
1	Examinations performed per ASME Code Case OMN-13, "Requirements for Extending Snubber Inservice Visual Examination Interval at LWR Power Plants."
2	Per the ASME OM Code, Subsection ISTD, 2004 Edition through the 2006 Addenda, Paragraph ISTD-5240 "Test Frequency."
3	Per the ASME OM Code, Subsection ISTD, 2004 Edition through the 2006 Addenda, Paragraph ISTD-4250 "Inservice Examination Intervals."

8.0 RELIEF REQUESTS FROM ASME SECTION XI

This section contains relief requests written per 10CFR50.55a(a)(3)(i) for situations where alternatives to ASME Section XI requirements provide an acceptable level of quality and safety; per 10CFR50.55a(a)(3)(ii) for situations where compliance with ASME Section XI requirements results in a hardship or an unusual difficulty without a compensating increase in the level of quality and safety; and per 10CFR50.55a(g)(5)(iii) for situations where ASME Section XI requirements are considered impractical.

The following NRC guidance was utilized to determine the correct 10CFR50.55a paragraph citing for DNPS relief requests. 10CFR50.55a(a)(3)(i) and 10CFR50.55a(a)(3)(ii) provide alternatives to the requirements of ASME Section XI, while 10CFR50.55a(g)(5)(iii) recognizes situational impracticalities.

10CFR50.55a(a)(3)(i): Cited in relief requests when alternatives to the ASME Section XI requirements which provide an acceptable level of quality and safety are proposed. Examples are relief requests which propose alternative non-destructive examination (NDE) methods and/or examination frequency.

10CFR50.55a(a)(3)(ii): Cited in relief requests when compliance with the ASME Section XI requirements is deemed to be a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Examples of hardship and/or unusual difficulty include, but are not limited to, excessive radiation exposure, disassembly of components solely to provide access for examinations, and development of sophisticated tooling that would result in only minimal increases in examination coverage.

10CFR50.55a(g)(5)(iii): Cited in relief requests when conformance with ASME Section XI requirements is deemed impractical. Examples of impractical requirements are situations where the component would have to be redesigned, or replaced to enable the required inspection to be performed.

An index for DNPS relief requests is included in Table 8.0-1. The "I5R-XX" relief requests are applicable to ISI, CISI, SPT, and PDI.

The following relief requests are subject to change throughout the inspection interval (e.g., NRC approval, withdrawal). Changes to NRC approved alternatives (other than withdrawal) require NRC approval.

TABLE 8.0-1
RELIEF REQUEST INDEX

Relief Request	Revision Date[3]	Status[2]	(Program) Description/ Approval Summary[1]
I5R-01	0 9/19/12	Submitted	**(ISI)** Inspection of Standby Liquid Control Nozzle Inner Radius. **Revision 0 Submitted.**
I5R-02	0 9/19/12	Submitted	**(ISI)** Alternate Risk-Informed Selection and Examination Criteria for Examination Categories B-F, B-J, C-F-1, and C-F-2 Pressure Retaining Piping Welds. **Revision 0 Submitted.**
I5R-03	0 9/19/12	Submitted	**(SPT)** Exemption From Pressure Testing Reactor Pressure Vessel Head Flange Seal Leak Detection System. **Revision 0 Submitted.**
I5R-04	0 9/19/12	Submitted	**(SPT)** Testing Frequency for Isolation Condenser Shell Side and Vent Piping. **Revision 0 Submitted.**
I5R-05	0 9/19/12	Submitted	**(SPT)** Continuous Pressure Monitoring of the Control Rod Drive (CRD) System Accumulators. **Revision 0 Submitted.**
I5R-06 (I4R-10, Rev. 1)	1 9/19/12	Authorized	**(ISI)** Extension of Relief for Alternative Reactor Pressure Vessel Circumferential Weld Examinations for Additional License Operating Period. **Revision 1 authorized per NRC SE dated 3/23/05. Permanent relief was authorized by NRC SE and thus applies to the 20 year extended period of operation of the renewed operating license, including this Fifth Inspection Interval.**
I5R-07	0 9/19/12	Submitted	**(ISI)** Use of BWRVIP Guidelines in Lieu of Specific ASME Code Requirements on Reactor Pressure Vessel Internals and Components Inspection. **Revision 0 Submitted.**
I5R-08		Reserved	

TABLE 8.0-1
RELIEF REQUEST INDEX

Relief Request	Revision Date[3]	Status[2]	(Program) Description/ Approval Summary[1]
I5R-09 **Exelon Fleet Relief Request**	0 9/19/12	Authorized	**(ISI)** Use of ASME Code Case N-789, Alternative Requirements for Pad Reinforcement of Class 2 and 3 Moderate Energy Carbon Steel Piping for Raw Water Service. **Revision 0 authorized per NRC SE dated 5/10/12, for both the Fourth and Fifth Intervals. (See Note 2 of the relief request for details of the NRC SE authorized change.)**
I5R-10	0 9/19/12	Submitted	**(ISI)** Expanded Applicability for Use of ASME Code Case N-532-4, Repair/Replacement Activity Documentation Requirements and Inservice Summary Report Preparation and Submission. **Revision 0 Submitted.**
I5R-11	0 9/19/12	Submitted	**(ISI)** Expanded Applicability for Use of ASME Code Case N-661-1, Alternative Requirements for Wall Thickness Restoration of Class 2 and 3 Carbon Steel Piping for Raw Water Service. **Revision 0 Submitted.**

Note 1: The NRC grants relief requests pursuant to 10CFR50.55a(g)(6)(i) when Code requirements cannot be met and proposed alternatives do not meet the criteria of 10CFR50.55(a)(3). The NRC authorizes relief requests pursuant to 10CFR50.55a(a)(3)(i) if the proposed alternatives would provide an acceptable level of quality and safety or under 10CFR50.55a(a)(3)(ii) if compliance with the specified requirements would result in hardship or unusual difficulties without a compensating increase in the level of safety.

Note 2: This column represents the status of the latest revision. Relief Request Status Options: Authorized - Approved for use in an NRC SE (See Note 1); Granted - Approved for use in an NRC SE (See Note 1); Authorized Conditionally - Approved for use in an NRC SE which imposes certain conditions; Granted Conditionally - Approved for use in an NRC SE which imposes certain conditions; Denied - Use denied in an NRC SE; Expired - Approval for relief has expired; Withdrawn - Relief has been withdrawn by the station; Not Required - The NRC has deemed the relief unnecessary in an SE or RAI; Cancelled - Relief has been cancelled by the station prior to issue; and Submitted - Relief has been submitted to the NRC by the station and is awaiting approval.

Note 3: The revision listed is the latest revision of the subject relief request. The date this revision became effective is the date of the approving SE which is listed in the fourth column of the table. The date noted in the second column is the date of the ISI Program Plan revision when the relief request was incorporated into the document.

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Request for Relief for Inspection of Standby Liquid Control Nozzle Inner Radius In Accordance with 10CFR50.55a(g)(5)(iii) Inservice Inspection Impracticality

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 Sht. 3
	Unit 3: ISI-125 Sht. 3

2. Applicable Code Edition and Addenda:

The Inservice Inspection Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2007 Edition through the 2008 Addenda.

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. Impracticality of Compliance:

Pursuant to 10CFR50.55a(g)(5)(iii), relief is requested on the basis that conformance with the Code requirements is impractical.

The Standby Liquid Control (SBLC) nozzle, as shown in Figure I5R-01.1, is designed with an integral socket to which the boron injection piping is fillet welded. The SBLC nozzle is located near the bottom of the vessel in an area which is inaccessible for ultrasonic examinations from the inside of the vessel. Therefore, ultrasonic examinations would need to be performed from the outside diameter of the vessel. As shown in Figure I5R-01.1, the

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ultrasonic scan would need to travel through the full thickness of the vessel into a complex cladding/socket configuration. These geometric and material reflectors inherent in the design prevent a meaningful examination from being performed on the inner radius of the SBLC nozzle.

In addition, the inner radius socket attaches to piping which injects boron at locations far removed from the nozzle. Therefore, the SBLC nozzle inner radius is not subjected to turbulent mixing conditions that are a concern at other nozzles.

5. Burden Caused By Compliance:

Compliance with the applicable Code requirements would require an ultrasonic examination to be performed on the outside diameter of the RPV. Geometric and material reflectors would prevent a meaningful examination, resulting in inaccurate data. Based on this, the Code requirements are deemed impractical in accordance with 10CFR50.55a(g)(5)(iii).

6. Proposed Alternate and Basis for Use:

As an alternate examination, Dresden Nuclear Power Station Units 2 and 3 will perform a VT-2 visual examination of the subject nozzles each refueling outage in conjunction with the Class 1 System Leakage Test.

7. Duration of Proposed Alternative:

Relief is requested for the Fifth Ten-Year Inspection Interval for Dresden Nuclear Power Station Units 2 and 3.

8. Precedents:

Dresden Nuclear Power Station Units 2 and 3 Fourth Inspection Interval Relief Request I4R-01 was authorized per NRC SE dated September 4, 2003. The Fifth Inspection Interval Relief Request utilizes a similar approach that was previously approved.

Quad Cities Nuclear Power Station Units 1 and 2 Fourth Inspection Interval Relief Request I4R-01 was authorized per NRC SE dated January 28, 2004.

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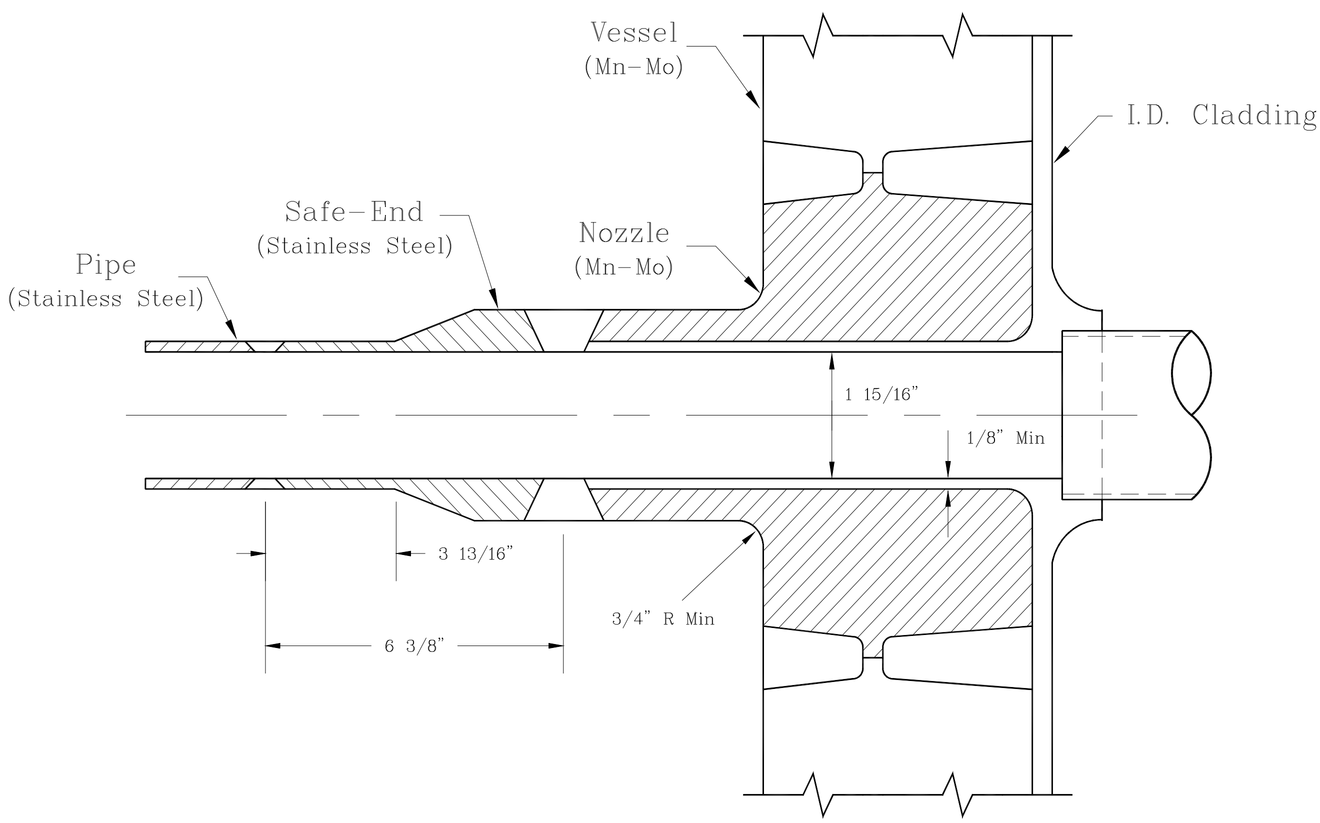
FIGURE I5R-01.1

2 INCH STANDBY LIQUID CONTROL NOZZLE



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Request for Relief for Alternate Risk-Informed Selection and Examination Criteria for Examination Categories B-F, B-J, C-F-1, and C-F-2 Pressure Retaining Piping Welds In Accordance with 10CFR50.55a(a)(3)(i) Alternative Provides Acceptable Level of Quality and Safety

1. ASME Code Component(s) Affected:

Code Class:	1 and 2
Examination Category:	B-F, B-J, C-F-1, and C-F-2
Item Number:	B5.10, B5.20, B9.11, B9.21, B9.31, B9.32, B9.40, C5.11, C5.41, C5.51, C5.70, and C5.81
Description:	Alternate Risk-Informed Selection and Examination Criteria for Examination Categories B-F, B-J, C-F-1, and C-F-2 Pressure Retaining Piping Welds
Component Number:	Pressure Retaining Piping

2. Applicable Code Edition and Addenda:

The Inservice Inspection Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2007 Edition through the 2008 Addenda.

3. Applicable Code Requirement:

Table IWB-2500-1, Examination Category B-F, requires volumetric and surface examinations on all welds for Item Numbers B5.10 and surface examinations on all welds for Item Number B5.20.

Table IWB-2500-1, Examination Category B-J, requires volumetric and surface examinations on a sample of welds for Item Numbers B9.11 and B9.31, and surface examinations on a sample of welds for Item Numbers B9.21, B9.32, and B9.40. The weld population selected for inspection includes the following:

1. All terminal ends in each pipe or branch run connected to vessels.

2. All terminal ends and joints in each pipe or branch run connected to other components where the stress levels exceed either of the following limits under loads associated with specific seismic events and operational conditions:

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a. primary plus secondary stress intensity range of $2.4S_m$ for ferritic steel and austenitic steel; and
b. cumulative usage factor U of 0.4.

3. All dissimilar metal welds not covered under Examination Category B-F.

4. Additional piping welds so that the total number of circumferential butt welds, branch connections, or socket welds selected for examination equals 25% of the circumferential butt welds, branch connection, or socket welds in the reactor coolant piping system. This total does not include welds excluded by IWB-1220.

Table IWC-2500-1, Examination Categories C-F-1 and C-F-2 require volumetric and surface examinations on a sample of welds for Item Numbers C5.11 and C5.51, and surface examinations on a sample of welds for Item Numbers C5.41, C5.70, and C5.81. The weld population selected for inspection includes the following:

1. Welds selected for examination shall include 7.5%, but not less than 28 welds, of all dissimilar metal, austenitic stainless steel and high alloy welds (Examination Category C-F-1) or of all carbon and low alloy steel welds (Examination Category C-F-2) not exempted by IWC-1220. (Some welds not exempted by IWC-1220 are not required to be nondestructively examined per Examination Categories C-F-1 and C-F-2. These welds, however, shall be included in the total weld count to which the 7.5% sampling rate is applied.) The examinations shall be distributed as follows:

   a. the examinations shall be distributed among the Class 2 systems prorated, to the degree practicable, on the number of nonexempt dissimilar metal, austenitic stainless steel and high alloy welds (Examination Category C-F-1) or carbon and low alloy welds (Examination Category C-F-2) in each system;

   b. within a system, the examinations shall be distributed among terminal ends, dissimilar metal welds, and structural discontinuities prorated, to the degree practicable, on the number of nonexempt terminal ends, dissimilar metal welds, and structural discontinuities in the system; and

   c. within each system, examinations shall be distributed between line sizes prorated to the degree practicable.

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4. Reason for Request:

Pursuant to 10CFR50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative utilizing Reference 1 along with two enhancements from Reference 4 will provide an acceptable level of quality and safety.

As stated in "Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)" (Reference 2):

> "*The staff concludes that the proposed RI-ISI Program as described in EPRI TR-112657, Revision B, is a sound technical approach and will provide an acceptable level of quality and safety pursuant to 10CFR50.55a for the proposed alternative to the piping ISI requirements with regard to the number of locations, locations of inspections, and methods of inspection.*"

The initial Dresden Nuclear Power Station Units 2 and 3 Risk-Informed Inservice Inspection (RISI) Program was submitted during the third period of the Third ISI Interval. This initial RISI Program was developed in accordance with EPRI TR-112657, Revision B-A, as supplemented by ASME Code Case N-578-1. The program was approved for use by the NRC via Safety Evaluation as transmitted to Exelon on September 5, 2001 (Reference 5).

The Dresden Nuclear Power Station RISI Program was resubmitted using the same approach during the Fourth ISI Interval for both Units 2 and 3. The program was approved for use by the NRC via Safety Evaluation as transmitted to Exelon on September 4, 2003 (Reference 6).

The transition from the 1995 Edition through the 1996 Addenda to the 2007 Edition through the 2008 Addenda of ASME Section XI for Dresden Nuclear Power Station's Fifth Interval does not impact the currently approved Risk-Informed ISI evaluation methods and process used in the Fourth ISI Interval, and the requirements of the new Code Edition/Addenda will be implemented as detailed in the Dresden Nuclear Power Station ISI Program Plan.

The Risk Impact Assessment completed as part of the original baseline RISI Program was an implementation/transition check on the initial impact of converting from a traditional ASME Section XI program to the new RISI methodology. For the Fifth Interval ISI update, there is no transition occurring between two different methodologies, but rather, the currently approved RISI methodology and evaluation will be maintained for the new interval. The original methodology of the evaluation has not changed, and

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the change in risk was simply re-assessed using the initial 1989 Edition, No Addenda ASME Section XI program prior to RISI and the new element selection for the Fifth Interval RISI Program. This same process has been maintained in each revision to the Dresden Nuclear Power Station RISI assessment that has been performed to date.

The actual "evaluation and ranking" procedure including the Consequence Evaluation and Degradation Mechanism Assessment processes of the currently approved (Reference 6) RISI Program remain unchanged and are continually applied to maintain the Risk Categorization and Element Selection methods of EPRI TR-112657, Revision B-A. These portions of the RISI Program have been and will continue to be reevaluated and revised as major revisions of the site Probabilistic Risk Assessment (PRA) occur and modifications to plant configuration are made. The Consequence Evaluation, Degradation Mechanism Assessment, Risk Ranking, Element Selection, and Risk Impact Assessment steps encompass the complete living program process applied under the Dresden Nuclear Power Station RISI Program.

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

The proposed alternative originally implemented in the "Risk Informed Inservice Inspection Plan, Dresden Units 2 and 3" (Reference 3), along with the two enhancements noted below, provide an acceptable level of quality and safety as required by 10CFR50.55a(a)(3)(i). This same program along with these enhancements was resubmitted and is currently approved for Dresden Nuclear Power Station's Fourth ISI Interval as documented in Reference 6.

The Fifth Inspection Interval RISI Program will be a continuation of the current application and will continue to be a living program as described in the Reason for Request section of this relief request. No changes to the evaluation methodology as currently implemented under EPRI TR-112657, Revision B-A, are required as part of this interval update. The following two enhancements will continue to be implemented.

a. In lieu of the evaluation and sample expansion requirements in Section 3.6.6.2, "RI-ISI Selected Examinations" of EPRI TR-112657, Dresden Nuclear Power Station will utilize the requirements of Paragraph -2430, "Additional Examinations" contained in ASME Code Case N-578-1 (Reference 4). The alternative criteria for additional examinations contained in ASME Code Case N-578-1 provide a more refined methodology for implementing necessary additional examinations. The reason for this selection is that the guidance discussed in EPRI TR-112657 includes requirements for additional examinations at a high level, based on service conditions, degradation mechanisms, and the performance of evaluations to determine the scope of additional examinations, whereas ASME Code Case N-578-1 provides more specific and clearer guidance

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regarding the requirements for additional examinations that is structured similar to the guidance provided in ASME Section XI, IWB-2430 and IWC-2430. Additionally, similar to the current requirements of ASME Section XI, Dresden Nuclear Power Station intends to perform additional examinations that are required due to the identification of flaws or relevant conditions exceeding the acceptance standards, during the outage the flaws are identified.

b. To supplement the requirements listed in Table 4-1, "Summary of Degradation-Specific Inspection Requirements and Examination Methods," of EPRI TR-112657, Dresden Nuclear Power Station will utilize the provisions listed in Table 1, Examination Category R-A, "Risk-Informed Piping Examinations," contained in ASME Code Case N-578-1 (Reference 4). To implement Note 10 of this table, paragraphs and figures from the 2007 Edition through the 2008 Addenda of ASME Section XI (Dresden Nuclear Power Station's Code of Record for the Fifth Interval) will be utilized which parallel those referenced in the Code Case for the 1989 Edition, No Addenda. Table 1 of ASME Code Case N-578-1 will be used as it provides a more detailed breakdown for "Examination Method" and "Categorization of Parts to be Examined." Based on these Methods and Categorization, the examination figures specified in Section 4 of EPRI TR-112657 will then be used to determine the examination volume/area based on the degradation mechanism and component configuration.

The Dresden Nuclear Power Station RISI Program, as developed in accordance with EPRI TR-112657, Rev. B-A (Reference 1), requires that 25% of the elements that are categorized as "High" risk (i.e., Risk Categories 1, 2, and 3) and 10% of the elements that are categorized as "Medium" risk (i.e., Risk Categories 4 and 5) be selected for inspection. For this application, the guidance for the examination volume for a given degradation mechanism is provided by the EPRI TR-112657 while the guidance for the examination method and categorization of parts to be examined are provided by the EPRI TR-112657 as supplemented by ASME Code Case N-578-1.

For NRC staff consideration in the evaluation of this alternative Risk-Informed ISI Program, Attachment 1 to the relief request contains a summary of the Regulatory Guide 1.200, Revision 2 (Reference 7), evaluation performed on Dresden Nuclear Station Quantification Notebook, DR PRA-014, Revision 2, December 22, 2009, (PRA Model DR209A) (Reference 8), and the impact of the identified gaps on the technical adequacy of the Dresden Nuclear Power Station PRA Model to support this RISI application (see Attachment 1, Table 1).

In addition to this risk-informed evaluation, selection, and examination procedure, all ASME Section XI piping components, regardless of risk classification, will continue to

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receive Code required pressure testing as part of the current ASME Section XI program. VT-2 visual examinations are scheduled in accordance with the Dresden Nuclear Power Station pressure testing program, which remains unaffected by the RISI Program.

6. Duration of Proposed Alternative:

Relief is requested for the Fifth Ten-Year Inspection Interval for Dresden Nuclear Power Station Units 2 and 3.

7. Precedents:

Dresden Nuclear Power Station Units 2 and 3 Fourth Inspection Interval Relief Request I4R-02 was authorized per NRC SE dated September 4, 2003. The Fifth Inspection Interval Relief Request utilizes a similar RISI methodology that was previously approved.

Three Mile Island Station Fourth Inspection Interval Relief Request I4R-02 was authorized per NRC SE dated July 20, 2011.

Clinton Power Station Third Inspection Interval Relief Request I3R-01 was authorized per NRC SE dated December 22, 2010.

8. References:

1) Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A, “Revised Risk-Informed Inservice Inspection Evaluation Procedure,” December 1999

2) Letter from W. H. Bateman (NRC) to G. L. Vine (EPRI) “Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999),” dated October 28, 1999

3) Initial Risk-Informed Inservice Inspection Evaluation – Dresden Nuclear Power Station Units 2 and 3, dated July 2000

4) American Society of Mechanical Engineers (ASME) Code Case N-578-1, “Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B”

5) Letter from A. J. Mendiola (NRC) to O. D. Kingsley (Exelon), “Safety Evaluation of Third Interval Risk-Informed Inservice Inspection Program Relief Request,” dated September 5, 2001

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6) Safety Evaluation from Anthony J. Mendiola (NRC) to John L. Skolds (Exelon), "Dresden Nuclear Power Station, Units 2 and 3 - Relief Request for Fourth 10-Year Inservice Inspection Interval (TAC Nos. MB6331, MB6332, MB7562, MB7563, MB7571, MB7572, MB7573, MB7574, MB7575, MB7576, MB7577, MB7578, MB7579, MB7580, MB7581, MB7582, MB7583, MB7584, MB7585, and MB7586)," dated September 4, 2003

7) Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated March 2009

8) Dresden Nuclear Power Station Quantification Notebook, DR PRA-014, Revision 2, December 22, 2009, PRA Model DR209A

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ATTACHMENT 1
DRESDEN 2009 PRA (DR209A) TECHNICAL CAPABILITY ASSESSMENT FOR RISK-INFORMED INSERVICE INSPECTION

Summary Statement of Dresden Nuclear Power Station PRA Model Capability for Use in Risk-Informed Inservice Inspection Applications

Introduction

Exelon Generation Company (EGC) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the Probabilistic Risk Assessment (PRA) models for all operating EGC nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the Dresden Nuclear Power Station PRA.

PRA Maintenance and Update

The EGC risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in the EGC Risk Management program, which consists of a governing procedure (ER-AA-600, "Risk Management") and subordinate implementation procedures. EGC procedure ER-AA-600-1015, "FPIE PRA Model Update," delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating EGC nuclear generation sites. The overall EGC Risk Management program, including ER-AA-600-1015, defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operating experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured, and their impact on CDF is trended.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities for equipment that can have a significant impact on the PRA model are updated approximately every four years.

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In addition to these activities, EGC risk management procedures provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events PRA models for EGC nuclear generation sites.
- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10CFR50.65(a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximately 4-year cycle; longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant.

The most recent update of the Dresden Nuclear Power Station PRA model (designated the DR209A model) (Reference 11) was completed in 2009 as a result of a regularly scheduled update to the previous 2005B PRA model. This model is the most recent evaluation of the risk profile at Dresden Nuclear Power Station for internal event challenges, including internal flooding. The Dresden Nuclear Power Station PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the Dresden Nuclear Power Station PRA is based on the event tree / fault tree methodology, which is a well-known methodology in the industry.

PRA Self Assessment and Peer Review

Several assessments of technical capability have been made and continue to be planned for the Dresden Nuclear Power Station PRA model. A chronological list of the assessments performed includes the following:

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- An independent PRA peer review was conducted under the auspices of the BWR Owners' Group (BWROG) in January 2001, following the Industry PRA Peer Review process (Reference 1). This peer review included an assessment of the PRA model maintenance and update process.
- In 2004, prior to the 2005 PRA update, a self-assessment analysis was performed against the available version of the ASME PRA Standard, Addendum A (Reference 2).
- During 2005 and 2006, the Dresden Nuclear Power Station PRA model results were evaluated in the BWR Owners' Group PRA cross-comparisons study performed in support of implementation of the Mitigating Systems Performance Indicator (MSPI) process (Reference 3).
- In March 2009, a focused PRA Peer Review was performed on the internal flooding element of the Dresden Nuclear Power Station 2005B full power internal events PRA. The results of the focused internal flood PRA Peer Review indicated a small number of the Internal Flood Supporting Requirements (SRs) were not met. The associated findings were added to the Updating Requirements Evaluation (URE) database to ensure resolution and have been included in the 2009 PRA self-assessment.
- In 2009, an update of the PRA self-assessment analysis was performed against ASME PRA Standard, Addendum B (Reference 4) following completion of the Dresden Nuclear Power Station 2009 PRA (DR209A) update. The self-assessment considered all of the findings from the 2009 focused PRA Peer Review on Internal Flooding. The 2009 self-assessment also addressed the updated Supporting Requirements associated with PRA Model Uncertainty as provided in the "Combined PRA Standard" (Reference 5).

A summary of the disposition of the 2001 Industry PRA Peer Review facts and observations (F&Os) for the Dresden Nuclear Power Station PRA models was documented as part of the statement of PRA capability for MSPI in the Dresden Nuclear Power Station MSPI Basis Document (Reference 3). As noted in that document, there were no significance level A F&Os from the peer review, and all significance level B F&Os were addressed and closed out with the completion of the previous Dresden Nuclear Power Station 2005B PRA model. Also noted in that submittal was the fact that, after allowing for plant-specific features, there were no MSPI cross-comparison outliers for Dresden Nuclear Power Station (refer to the third bulleted item above).

A Self-Assessment (Gap Analysis) for the Dresden Nuclear Power Station PRA model was completed in 2004 in preparation for the 2005 PRA update. This Gap Analysis was performed

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against the ASME PRA Standard (Reference 2). This gap analysis defined a list of 94 Supporting Requirements from the Standard for which potential gaps to Capability Category II of the Standard were identified. For each such potential gap, a PRA URE (EGC model update tracking database) was documented for resolution.

A previous PRA model update was completed in 2005. In updating the PRA, changes were made to the PRA to address most of the identified gaps, as well as to address other open UREs. Following the update, an assessment of the status of the gap analysis relative to the new model and the updated requirements in Addendum B of the ASME PRA Standard concluded that 69 of the gaps were fully resolved (i.e., are no longer gaps), and one (1) was partially resolved. After accounting for the number of Supporting Requirements that were added or deleted as part of Addendum B, the Dresden Nuclear Power Station PRA contained 25 gaps to Capability Category II of the Standard.

As indicated above, a recent PRA model update was completed in 2009, resulting in the DR209A updated PRA model. In updating the PRA, changes were made to address most of the previously identified gaps as well as to address other open UREs. Following the update, an assessment of the status of the Gap Analysis relative to the new PRA model and the requirements in Addendum B of the ASME PRA Standard (Reference 4) concluded that 22 of the gaps were fully resolved (i.e., are no longer gaps). The Dresden Nuclear Power Station 2009 PRA (DR209A) contains 3 potential gaps to Capability Category II of the Standard (Reference 4).

A summary of this assessment of the current open items relative to the RISI relief request is provided in Table 1. All remaining gaps will be reviewed for consideration during the next Dresden Nuclear Power Station model update (anticipated to be 2013) but are judged to have low impact on the PRA model or its ability to support a full range of PRA applications. These items are or are being documented in the PRA URE database so that they can be tracked and their potential impacts accounted for in applications where appropriate. In addition, plant changes made since the last PRA update have been reviewed and determined to not have a significant PRA impact. These items are also documented in UREs for consideration in future PRA updates, as appropriate.

Guidance from EPRI Report on PRA Technical Adequacy for RISI

EPRI report TR-1021467-A (Reference 9) provides guidance on the PRA Standard Capability Category necessary to support RISI. This report received a Safety Evaluation (SE) from the Nuclear Regulatory Commission (NRC) in January 2012. Reg. Guide 1.200 considers it a good

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practice to have, in general, SRs meet Capability Category II for applications. However, according to the EPRI report not all SRs require Capability Category II to adequately support RISI applications. According to the EPRI report the Dresden Nuclear Power Station gaps listed in Table 1 (SC-A5, DA-C8, and DA-C12) do not require Capability Category II, but instead only require Capability Category I, the most basic level. Therefore according to EPRI TR-1021467-A and the associated NRC SE, the Dresden Nuclear Power Station PRA model DR209A is adequate for use in the RISI application.

Sensitivity Studies

Three gaps to the PRA standard were identified in Table 1 (SC-A5, DA-C8, and DA-C12). These gaps have been reviewed to determine which, if any, would merit RISI-specific sensitivity studies in the presentation of the application results. The result of this assessment concluded that no additional sensitivity studies are merited.

General Conclusion Regarding PRA Capability

The Dresden Nuclear Power Station PRA maintenance and update processes and technical capability evaluations described above provide a robust basis for concluding that the PRA is suitable for use in risk-informed inservice inspection applications. As specific risk-informed PRA applications are performed, remaining gaps to specific requirements in the PRA standard will be reviewed with EPRI technical adequacy guidance (Reference 9), when applicable, to determine which, if any, would merit application-specific sensitivity studies in the presentation of the application results.

Conclusion Regarding PRA Capability for Risk-Informed ISI

The Dresden Nuclear Power Station PRA model continues to be suitable for use in the risk-informed inservice inspection application. This conclusion is based on:

- PRA maintenance and update processes in place.
- PRA technical capability evaluations that have been performed and are being planned.
- RISI process considerations, as noted above, that demonstrate the relatively limited sensitivity of the EPRI RISI process to PRA attribute capability beyond ASME PRA Standard Capability Category I.

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In support of the PRA analyses for the Dresden Nuclear Power Station 10-year interval evaluation using the Dresden Nuclear Power Station DR209A PRA model, the remaining gaps to the PRA standard have been reviewed to determine which, if any, would merit RISI-specific sensitivity studies in the presentation of the application results. The result of this assessment concluded that no additional sensitivity studies are merited.

References

1. Boiling Water Reactors Owners' Group, *BWROG PSA Peer Review Certification Implementation Guidelines*, Revision 3, January 1997

2. American Society of Mechanical Engineers, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, (ASME RA-S-2002), Addenda RA-Sa-2003, December 2003

3. Dresden Nuclear Power Station MSPI Basis Document, DR-MSPI-001, Revision 5, September 2009

4. American Society of Mechanical Engineers, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, (ASME RA-S-2002), Addenda RA-Sb-2005, December 2005

5. ASME Committee on Nuclear Risk Management in collaboration with ANS Risk Informed Standards Committee, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications*, ASME/ANS RA-Sa-2009, March 2009

6. Not used

7. U.S. Nuclear Regulatory Commission, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, Regulatory Guide 1.174, Revision 1, November 2002

8. Regulatory Guide 1.200, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities*, Revision 2, March 2009

9. *Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-informed Inservice Inspection Programs*, EPRI TR-1021467-A, June 2012

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ATTACHMENT 1
DRESDEN 2009 PRA (DR209A) TECHNICAL CAPABILITY ASSESSMENT FOR RISK-INFORMED INSERVICE INSPECTION

10. Self-Assessment of the Dresden Nuclear Power Station PRA Against the ASME PRA Standard Requirements, DR PSA-016, Revision 2, October 2009

11. Dresden Nuclear Power Station Quantification Notebook, DR PRA-014, Revision 2, December 2009

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ATTACHMENT 1
DRESDEN PRA MODEL DR209A TECHNICAL CAPABILITY ASSESSMENT FOR RISK-INFORMED INSERVICE INSPECTION

TABLE 1 – Status of Identified Gaps to Capability Category II of the ASME PRA Standard

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RISI
Gap #1	Mission times are discussed in Success Criteria (DR-PSA-003). The mission times for failure to run calculations are assessed at 24 hours or less if specifically justified. Extending the FTR mission time beyond 24 hours for loss of DHR sequences is considered to be an unnecessary complication and does not affect PRA insights nor does it significantly affect its quantitative evaluation. The evaluation of safe stable states in a PSA has generally involved the assessment of equipment operation and operator actions over an extended period of time. This extended period of time is nominally taken to be sufficiently long such that offsite resources can be brought to bear to mitigate or further prevent accident progression. The considerations that have dominated the choice of the mission time are as follows: • Equipment failure rates (failures/hour) are judged to be too conservative for times greater than a few hours of operation. • For times greater than a few hours, the ability to repair and recover equipment can compete with the failure rate such that there can be considered to be a steady state equilibrium condition reached. • For times greater than 24 hours, the TSC and EOF would be manned, and additional expertise could be available by phone or transported to these facilities. • For times greater than 24 hours, it is considered highly likely that offsite resources (e.g., equipment, power, vehicles) would be available as back-ups to primary methods of prevention and mitigation.	SC-A5	Open. Enhance documentation to justify why extending FTR mission times beyond 24 hours for loss of DHR sequences is not necessary. The considerations that support the choice of the mission time are as follows: Equipment failure rates (failures/hour) are judged to be too conservative for times greater than a few hours of operation. For times greater than a few hours, the ability to repair and recover equipment can compete with the failure rate such that there can be considered to be a steady state equilibrium condition reached.	The current approach is judged to be reasonable for long term scenarios (e.g., long term loss of DHR) and is adequate for RISI application.

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ATTACHMENT 1
DRESDEN PRA MODEL DR209A TECHNICAL CAPABILITY ASSESSMENT FOR RISK-INFORMED INSERVICE INSPECTION

TABLE 1 – Status of Identified Gaps to Capability Category II of the ASME PRA Standard

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RISI
	• From a risk perspective, actual data from natural and man-caused disasters have indicated that public evacuations can be effectively carried out in time frames of less than 24 hours. Therefore, prevention of accidents through 24 hours of mission time have the largest potential for early health effects risk reduction. • Finally, beyond time frames of 24 hours, "ad hoc" procedures can be written and reviewed to perform alignments and equipment usage that are not part of current plant practices or training. Such ad hoc procedures and equipment usage can cover such a wide spectrum of possibilities that it is judged not useful to develop all possible contingencies at this time. Based on the above considerations, it has been considered in past PSAs that it is to appropriate to use an equipment mission time of 24 hours. This consideration dictates the use of equipment "run" failure rates (per hour) coupled with a 24 hour mission time to calculate the "run" failure probability of equipment. This calculated "run" failure probability is then treated conservatively by applying this "run" failure probability as a failure that is postulated at time zero.			
Gap #2	Plant specific operational records were not used to quantify the time systems or trains were in standby. A detailed determination is judged to require a significant level of resources with marginal quantitative benefit. An estimate of the time that components were in standby is judged to be sufficient.	DA-C8	Open.	The PRA model is judged to appropriately estimate the time that components were in standby for calculating the standby failure rate, and therefore is adequate for the RISI application.

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ATTACHMENT 1
DRESDEN PRA MODEL DR209A TECHNICAL CAPABILITY ASSESSMENT FOR RISK-INFORMED INSERVICE INSPECTION

TABLE 1 – Status of Identified Gaps to Capability Category II of the ASME PRA Standard

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RISI
Gap #3	No interviews of plant staff were performed to generate uncertainty estimates of unavailability per maintenance act. An exception is taken to DA-C12. The plant staff does not have reasonable insights applicable to the level of uncertainty associated with the maintenance durations. Most plant staff have rotated positions and do not have sufficient longevity to provide this insight.	DA-C12	Open.	The model is consistent with data from the plant MR database, which is adequate for RISI application.

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Request for Relief for Exemption from Pressure Testing Reactor Pressure Vessel Head Flange Seal Leak Detection System In Accordance with 10CFR50.55a(a)(3)(ii) Hardship or Unusual Difficulty Without Compensating Increase in Level of Quality and Safety

1. ASME Code Component(s) Affected:

Code Class: 2
Reference: IWC-2500, Table IWC-2500-1
Examination Category: C-H
Item Number: C7.10
Description: Exemption From Pressure Testing Reactor Pressure Vessel Head Flange Seal Leak Detection System
Component Number:

Unit No.	Drawing	Test Block No.
Unit 2	M-26 Sh. 1	2NB01
Unit 3	M-357 Sh. 1	3NB01

2. Applicable Code Edition and Addenda:

The Inservice Inspection Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2007 Edition through the 2008 Addenda.

3. Applicable Code Requirements:

Table IWC-2500-1, Examination Category C-H, Item Number C7.10, requires all Class 2 pressure retaining components be subject to a system leakage test with a VT-2 visual examination in accordance with IWC-5220. This pressure test is to be conducted once each inspection period.

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4. Reason for Request:

Pursuant to 10CFR50.55a(a)(3)(ii), relief is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The Reactor Pressure Vessel Head Flange Leak Detection Line is separated from the reactor pressure boundary by one passive membrane, a silver plated O-ring located on the vessel flange. A second O-ring is located on the opposite side of the tap in the vessel flange (see Figure I5R-03.2). This line is required during plant operation in order to indicate failure of the inner flange seal O-ring. Failure of the O-ring would result in the annunciation of a High Level Alarm in the control room. On this annunciation, control room operators would quantify the leakage rate from the O-ring and then isolate the leak detection line from the drywell sump by closing the AO 2(3)-220-51 valve (see Figure I5R-03.1). This action is taken in order to prevent steam cutting of the O-ring and the vessel flange. Failure of the inner O-ring is the only condition under which this line is pressurized.

The configuration of this system precludes manual testing while the vessel head is removed because the odd configuration of the vessel tap (see Figure I5R-03.2) coupled with the high test pressure requirement (1000 psig minimum), prevents the tap in the flange from being temporarily plugged. The opening in the flange is only 3/16 of an inch in diameter and is smooth walled making a high pressure temporary seal very difficult. Failure of this seal could possibly cause ejection of the device used for plugging into the vessel.

A pneumatic test performed with the head installed is precluded due to the configuration of the top head. The top head of the vessel contains two grooves that hold the O-rings. The O-rings are held in place by a series of retainer clips spaced 15^o apart. The retainer clips are contained in a recessed cavity in the top head (see Figure I5R-03.3). If a pressure test was performed with the head on, the inner O-ring would be pressurized in a direction opposite to what it would see in normal operation. This test pressure would result in a net inward force on the O-ring that would tend to push it into the recessed cavity that houses the retainer clips. The O-ring material is only .050" thick with a silver plating thickness of .004" to .006" and could very likely be damaged by this deformation into the recessed areas on the top head.

In addition to the problems associated with the O-ring design that preclude this testing it is also questionable whether a pneumatic test is appropriate for this line. Although the line will initially contain steam if the inner O-ring leaks, the system actually detects leakage rate by measuring the level of condensate in a collection chamber. This would make the system medium water at the level switch. Finally, the use of a pneumatic test performed at a minimum of 1000 psig would represent an unnecessary risk in safety for the inspectors and test engineers in the unlikely event of a test failure, due to the large amount of stored energy contained in air pressurized to 1000 psig.

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System leakage testing of this line is precluded because the line will only be pressurized in the event of a failure of the inner O-ring. It is extremely impractical to purposely fail the inner O-ring in order to perform a test.

Based on the above, Dresden Nuclear Power Station Units 2 and 3 requests relief from the ASME Section XI requirements for system leakage testing of the Reactor Pressure Vessel Head Flange Seal Leak Detection System.

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

A VT-2 visual examination will be performed on the line during vessel flood-up in a refueling outage. The static head developed due to the water above the vessel flange during flood-up will allow for the detection of any gross indications in the line. This examination will be performed with the frequency specified by Table IWC-2500-1 for a System Leakage Test (once each inspection period).

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

Relief is requested for the Fifth Ten-Year Inspection Interval for Dresden Nuclear Power Station Units 2 and 3.

7. <u>Precedents:</u>

Dresden Nuclear Power Station Units 2 and 3 Fourth Inspection Interval Relief Request I4R-05 was granted per NRC SE dated September 4, 2003. The Fifth Inspection Interval Relief Request utilizes a similar approach that was previously approved.

Three Mile Island Station Fourth Inspection Interval Relief Request I4R-03 was granted per NRC SE dated July 20, 2011.

Clinton Power Station Third Inspection Interval Relief Request I3R-03 was authorized per NRC SE dated December 22, 2010.

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FIGURE I5R-03.1

HEAD FLANGE SEAL LEAK DETECTION SCHEMATIC

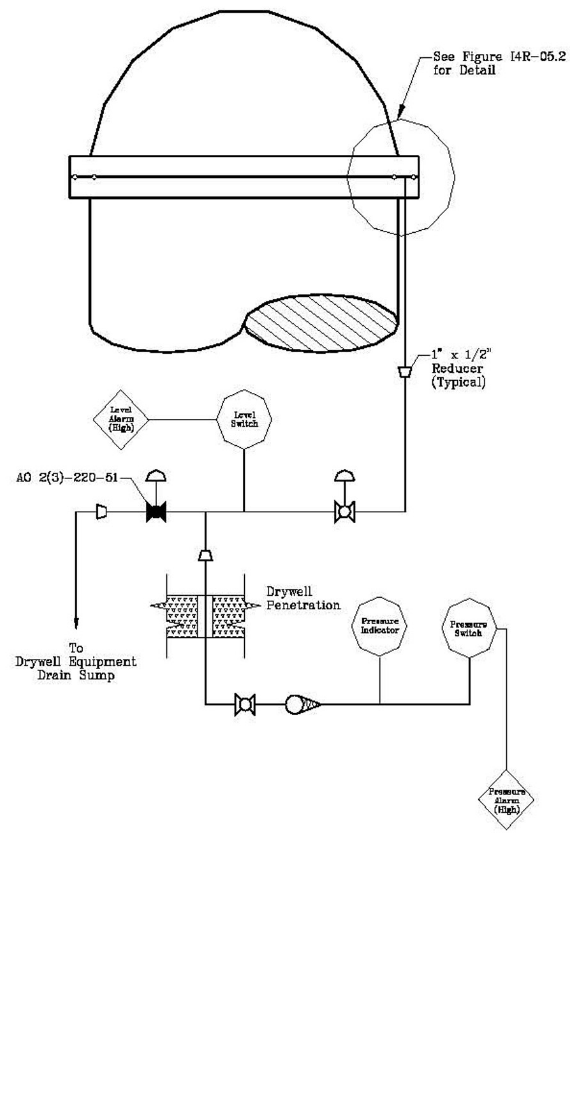

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FIGURE I5R-03.2

FLANGE SEAL LEAK DETECTION LINE DETAIL

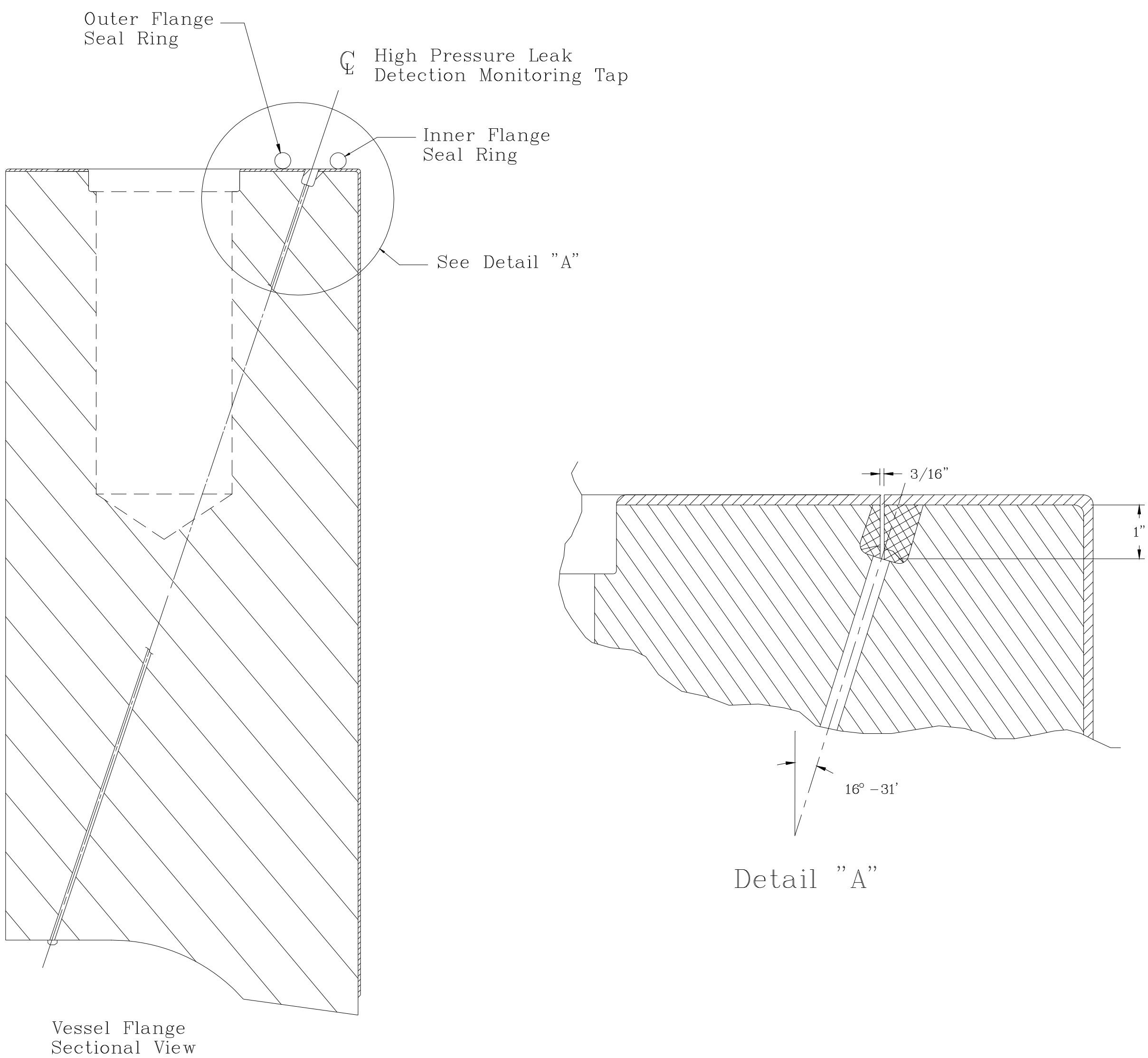

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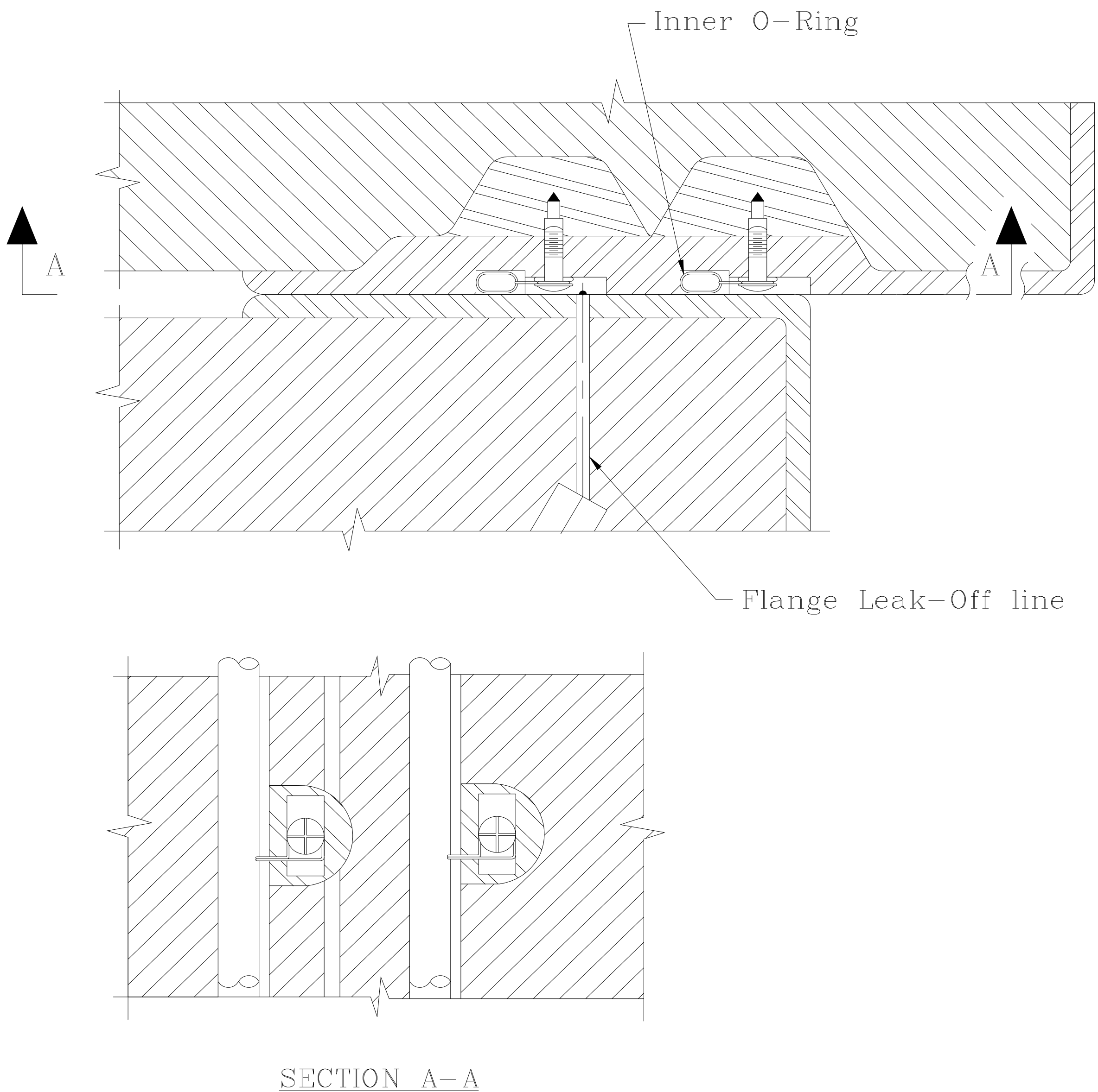
FIGURE I5R-03.3

VESSEL TOP HEAD O-RING RETAINER DETAIL



10CFR50.55a RELIEF REQUEST I5R-04
Revision 0

Request for Relief for Testing Frequency for Isolation Condenser Shell Side and Vent Piping In Accordance with 10CFR50.55a(a)(3)(i) Alternative Provides Acceptable Level of Quality and Safety

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 Vent Piping
Component Number:	

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

2. Applicable Code Edition and Addenda:

The Inservice Inspection Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2007 Edition through the 2008 Addenda.

3. Applicable Code Requirement:

IWD-5222(b) states that in the case of atmospheric storage tanks, the nominal hydrostatic head, developed with the tank filled to its design capacity shall be acceptable as the system test pressure.

Table IWD-2500-1, 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. This pressure test is to be conducted once each inspection period.

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Revision 0

4. Reason for Request:

Pursuant to 10CFR50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative provides an acceptable level of quality and safety.

The Isolation Condenser does not have a design level; instead, it has a design pressure. The design pressure of the shell side of the Isolation Condenser is 25 psig. It is impossible to develop this pressure in the Isolation Condenser shell side because the condenser is vented to the atmosphere through a nonisolable line and the condenser is only 12 ft in height. A hydrostatic head of 25 psig would correspond to approximately 58 ft of water.

Although the shell side of the condenser is designed for 25 psig, the system is normally operated with the Isolation Condenser water at a level greater than or equal to 6 ft. It would be an abnormal activity and would be impractical to fill the Isolation Condenser to the top simply to achieve a few more pounds of static head pressure. This water would have to be drained and processed by radwaste.

The 3.0 psig difference in test pressure (with the Isolation Condenser filled to the top vs. the minimum operating level) is so slight that a test with the Isolation Condenser filled to the normal operating level is essentially the same as a test with the Isolation Condenser filled to the top. Once every five years, Dresden Nuclear Power Station Units 2 and 3 performs a normal operational test of the isolation condenser in accordance with the plant Technical Requirements Manual. This test adequately challenges the isolation condenser and associated piping to allow inspectors to conduct a VT-2 visual examination that meets the System Leakage Test requirements of Table IWD-2500-1. However, since this test is only performed once every five years, the requirement to conduct a VT-2 visual examination once per period cannot be met.

To complete the interval inspection requirements of the isolation condenser and associated piping, normal static conditions (filled greater than or equal to 6 ft per the discussion above) exist under which a VT-2 visual examination can be performed.

5. Proposed Alternate and Basis for Use:

The Isolation Condenser and associated piping in Test Blocks 2(3)IC01 and 2(3)IC02 will be VT-2 visually examined during the Technical Requirements Manual 5-year operational test. This test will cover two of the three inservice inspection periods. Additionally, Test Block 2(3)IC02 will be VT-2 visually examined under normal static conditions (filled greater than or equal to 6 ft) during the remaining period in which the normal operational test is not performed. These pressure tests will be performed as alternatives to the periodic and interval testing requirements of Table IWD-2500-1.

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

Relief is requested for the Fifth Ten-Year Inspection Interval for Dresden Nuclear Power Station Units 2 and 3.

7. <u>Precedents:</u>

Dresden Nuclear Power Station Units 2 and 3 Fourth Inspection Interval Relief Request I4R-06 was authorized per NRC SE dated September 4, 2003. The Fifth Inspection Interval Relief Request utilizes a similar approach that was previously approved.

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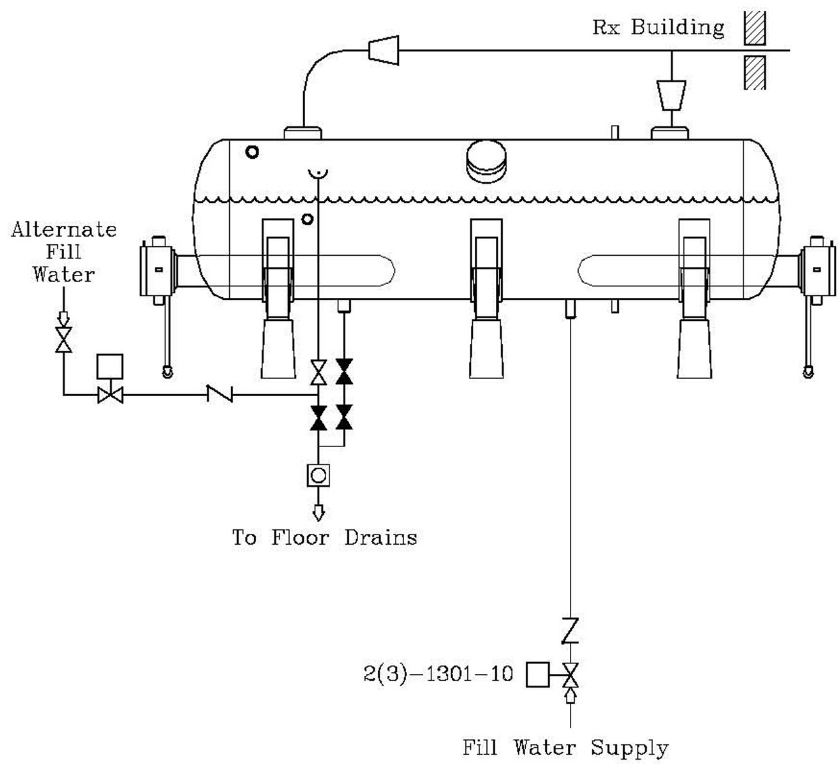
FIGURE I5R-04.1

ISOLATION CONDENSER SYSTEM SIMPLIFIED SCHEMATIC



10CFR50.55a RELIEF REQUEST I5R-05
Revision 0

Request for Relief for Continuous Pressure Monitoring of the Control Rod Drive (CRD) System Accumulators In Accordance with 10CFR50.55a(a)(3)(i) Alternative Provides Acceptable Level of Quality and Safety

1. ASME Code Component(s) Affected:

Code Class: 2
Reference: IWC-2500, Table IWC-2500-1
Examination Category: C-H
Item Number: C7.10
Description: Continuous Pressure Monitoring of the Control Rod Drive (CRD) System Accumulators
Component Number:

Unit No.	Drawing	Test Block No.
Unit 2	M-34 Sh. 2	2RC02
Unit 3	M-365 Sh. 2	3RC02

2. Applicable Code Edition and Addenda:

The Inservice Inspection Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2007 Edition through the 2008 Addenda.

3. Applicable Code Requirement:

Table IWC-2500-1, Examination Category C-H, Item Number C7.10, requires all Class 2 pressure retaining components be subject to a system leakage test with a VT-2 visual examination in accordance with IWC-5220. This pressure test is to be conducted once each inspection period.

4. Reason for Request:

Pursuant to 10CFR50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative provides an acceptable level of quality and safety.

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As required by Dresden Nuclear Power Station Units 2 and 3 Technical Specifications, the CRD system accumulator pressure must be greater than or equal to 940 psig to be considered operable. The accumulator pressure is continuously monitored by system instrumentation. Since the accumulators are isolated from the source of make-up nitrogen, the continuous monitoring of the CRD accumulators functions as a pressure decay type test. Should accumulator pressure fall below 1000 psig, an alarm is received in the control room. The pressure drop for the associated accumulator is then recorded, and the accumulator is recharged in accordance with Dresden Nuclear Power Station procedures. If an accumulator requires charging more than twice in a thirty day period, then a leak check is performed to determine the cause of the pressure loss. When leakage is detected, corrective actions are taken to repair the leaking component as required by Dresden Nuclear Power Station procedures.

Since monitoring the nitrogen side of the accumulators is continuous, any leakage from the accumulator would be detected by normal system instrumentation. An additional VT-2 visual examination performed once per inspection period would not provide an increase in safety, system reliability, or structural integrity. In addition, performance of a VT-2 visual examination would require applying a leak detection solution to 177 accumulators per unit resulting in additional radiation exposure without any added benefit in safety. This inspection would thus not be consistent with As Low As Reasonably Achievable (ALARA) practices.

Relief is requested from the VT-2 visual examination requirements specified in Table IWC-2500-1 for the nitrogen side of the CRD system accumulators on the basis that Dresden Nuclear Power Station Technical Specification Surveillance requirements exceed the code requirement for a VT-2 visual examination.

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

As an alternate to the VT-2 visual examination requirements of Table IWC-2500-1, Dresden Nuclear Power Station will perform continuous pressure decay monitoring in conjunction with Technical Specifications for the nitrogen side of the CRD accumulators including attached piping.

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

Relief is requested for the Fifth Ten-Year Inspection Interval for Dresden Nuclear Power Station Units 2 and 3.

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7. Precedents:

Dresden Nuclear Power Station Units 2 and 3 Fourth Inspection Interval Relief Request I4R-07 was authorized per NRC SE dated September 4, 2003. The Fifth Inspection Interval Relief Request utilizes a similar approach that was previously approved.

LaSalle County Nuclear Power Station Units 1 and 2 Third Inspection Interval Relief Request I3R-09 was authorized per NRC SE dated January 30, 2008. Revision 1 submitted under Supplement RA-07-036a, dated July 20, 2007.

Quad Cities Nuclear Power Station Units 1 and 2 Fourth Inspection Interval Relief Request I4R-06 was authorized per NRC SE dated January 28, 2004.

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Request for Relief for Extension of Relief for Alternative Reactor Pressure Vessel Circumferential Weld Examinations for Additional License Operating Period In Accordance with 10CFR50.55a(a)(3)(i) Alternative Provides Acceptable Level of Quality and Safety

*** NOTE ***

Dresden Nuclear Power Station Units 2 and 3 Fifth Inspection Interval Relief Request I5R-06, Revision 1 is provided for completeness and has no technical changes from the previously approved relief request. This relief request was previously submitted to address both Dresden Nuclear Power Station and Quad Cities Nuclear Power Station on February 23, 2004, and approved under the Fourth Inspection Interval ISI Program Plan as Relief Request I4R-10, Revision 1. The approval authorized under NRC SE dated March 23, 2005, for Dresden Nuclear Power Station, was for permanent relief and thus applies to the 20 year extended period of operation of the renewed operating license, including this Fifth Inspection Interval.

The relief request is carried here and renumbered as I5R-06, Revision 1. All ASME Code references were made in accordance with the 1995 Edition through the 1996 Addenda of ASME Section XI. No changes to the actual approved relief request have been made and no further or revised authorization is required.

1. ASME Component(s) Affected:

Components affected are American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Class 1 pressure retaining reactor pressure vessel (RPV) shell circumferential welds, Examination Category B-A, Item Number B1.11.

2. Applicable Code Edition and Addenda:

The applicable ASME Code, Section XI, for Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, is the 1995 Edition through the 1996 Addenda.

3. Applicable Code Requirement:

In accordance with the provisions of 10CFR50.55a, "Codes and standards," paragraph (a)(3)(i), Exelon Generation Company, LLC (EGC) requests permanent relief for the additional license operating period requested in Reference 1 for DNPS, Units 2 and 3,

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and QCNPS, Units 1 and 2, from the requirement of ASME Code, Section XI, Subarticle IWB-2500, Table IWB-2500-1, Examination Category B-A, Item Number B1.11.

Subarticle IWB-2500 requires components specified in Table IWB-2500-1 to be examined. Table IWB-2500-1 requires volumetric examination of all RPV shell circumferential welds each inspection interval (i.e., Examination Category B-A, Item Number B1.11).

4. Reason for Request:

Reference 2 provides the technical basis for permanently deferring the augmented inspections of circumferential welds in boiling water reactor (BWR) RPVs. In the report, the BWR Vessel and Internals Project (BWRVIP) concluded that the probabilities of failure for BWR RPV circumferential welds are orders of magnitude lower than that of the longitudinal welds. The NRC conducted an independent risk-informed, probabilistic fracture mechanics assessment (PFMA) of the analysis presented in Reference 2, and the results are documented in Reference 3. EGC has determined that the proposed alternative described below provides an acceptable level of quality and safety and satisfies the requirements of 10CFR50.55a(a)(3)(i).

5. Proposed Alternative and Basis for Use:

Proposed Alternative

In accordance with 10CFR50.55a(a)(3)(i), and consistent with information contained in Reference 4, EGC proposes the following alternate provisions for the subject weld examinations since the proposed alternative provides an acceptable level of quality and safety.

The failure frequency for RPV shell circumferential welds is sufficiently low to justify their elimination from the Inservice Inspection (ISI) requirement of ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item Number B1.11.

The ISI examination requirements of the ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item Number B1.12, RPV shell longitudinal welds (i.e., also known as vertical or axial welds) shall be performed, to the extent possible, and shall include inspection of the circumferential welds only at the intersection of these welds with the longitudinal welds, or approximately 2 to 3 percent of the RPV shell circumferential welds. When this examination is performed, an automated ultrasonic inspection system will provide the best possible examination of the RPV shell longitudinal welds. These welds are generally only accessible from inside surfaces of the RPV using an automated ultrasonic inspection system, which provides the best possible

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Revision 1

examination of the RPV shell longitudinal welds. Inspections from the outside surfaces have limited access due to the close proximity of the biological shield to the RPV. Also, the reflective insulation that occupies this space is not designed for removal.

Basis for Use

Reference 2 provides the technical basis to justify relief from the examination requirements of RPV shell circumferential welds. The results of the NRC's evaluation of Reference 2 are documented in Reference 3. Reference 4 permits BWR licensees to request permanent (i.e., for the remaining term of operation under the existing, initial, license) relief from the ISI requirements of 10CFR50.55a(g) for the volumetric examination of RPV shell circumferential welds (i.e., ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item Number B1.11). This relief can be granted by demonstrating that:

1. at the expiration of their license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staff's July 30, 1998, safety evaluation, and

2. licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the staff's July 30, 1998, safety evaluation.

Generic Letter 98-05, Criterion 1

Demonstrate that at the expiration of their license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the NRC's July 28, 1998, safety evaluation.

Response

The NRC evaluation of BWRVIP-05 utilized the FAVOR code to perform a PFMA to estimate the RPV shell weld failure probabilities. Three key assumptions of the PFMA are: (1) the neutron fluence used was the estimated end-of-life mean fluence, (2) the chemistry values are mean values based on vessel types, and (3) the potential for beyond-design-basis events is considered.

Tables 1 and 2 provide a comparison of the limiting RPV circumferential weld parameters for each DNPS and QCNPS unit to those found in Table 2.6-5 of the NRC final safety evaluation of BWRVIP-05 (i.e., Reference 3) for a Babcock and Wilcox vessel. Although the chemistry composition and chemistry factor for DNPS Unit 3 are higher than the limits of the NRC analysis, the shifts in reference temperature for both

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units are lower than the shift from the NRC limiting analysis. In addition, the unirradiated reference temperatures for both DNPS units are lower. The combination of unirradiated reference temperature and embrittlement shift yields adjusted reference temperatures considerably lower than the NRC mean analysis values.

The chemistry composition and chemistry factor for QCNPS Unit 1 are less than or equal to the limits of the NRC analysis. While the nickel content for Unit 2 is higher than the value utilized in the NRC analysis, the Unit 2 copper content and the chemistry factor are considerably lower than the values utilized in the NRC analysis. Additionally, the unirradiated reference temperatures for both QCNPS units are lower than the NRC limits. The combination of unirradiated reference temperature and embrittlement shift yields adjusted reference temperatures considerably lower than the NRC mean analysis values.

The end of life (i.e., 54 effective full power year (EFPY)) inside diameter fluences for DNPS, Units 2 and 3, and QCNPS, Units 1 and 2, are considerably lower than the NRC estimated 54 EFPY fluence. The 54 EFPY fluence estimates were calculated using the fluence methodology of General Electric Nuclear Energy licensing topical report NEDC-32983P (i.e., Reference 5), which was approved by the NRC in Reference 6, and adheres to the guidance of Regulatory Guide 1.190 (i.e., Reference 7). The end of extended license operating time of 54 EFPY includes the extended power uprate approved by the NRC in References 8 and 9. There are no additional uprates planned for DNPS or QCNPS.

The shifts in reference temperature for all four units are lower than the 54 EFPY shift from the NRC analysis. Therefore, for each unit, the RPV shell weld embrittlement due to fluence is calculated to be less than the NRC's limiting case, and each unit's RPV shell circumferential weld failure probabilities are bounded by the conditional failure probability, P(FIE), in the NRC's limiting plant specific analysis (54 EFPY) through the projected additional license operating period. For these reasons, the DNPS, Units 2 and 3, and QCNPS, Units 1 and 2, RPVs are bounded by Reference 3.

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Table 1: Effects of Irradiation on RPV Circumferential Weld Properties – DNPS			
Parameter Description	**DNPS Unit 2 Parameters at 54 EFPY (Weld Wire Heat/Flux Lot # 71249/8504)**	**DNPS Unit 3 Parameters at 54 EFPY (Weld Wire Heat/Flux Lot # 299L44/8650 (WF19/WF-25))**	**NRC Limiting Plant Specific Analysis (54 EFPY)**
Copper (weight %)	0.23	0.34	0.31
Nickel (weight %)	0.59	0.68	0.59
Chemistry Factor	168	221	196.7
End of Life Inside Diameter Fluence (10^{19} n/cm^2)	0.042	0.041	0.19
ΔRT_{NDT} (°F)	44	58	109.4
$\Delta RT_{NDT(U)}$ (°F)	10	-5	20
Mean RT_{NDT} (°F)	54	53	129.4

Table 2: Effects of Irradiation on RPV Circumferential Weld Properties – QCNPS			
Parameter Description	**QCNPS Unit 1 Parameters at 54 EFPY (Weld Wire Heat/Flux Lot # 406L44/8688)**	**QCNPS Unit 2 Parameters at 54 EFPY (Weld Wire Heat/Flux Lot # S3986/3870) Linde 124**	**NRC Limiting Plant Specific Analysis (54 EFPY)**
Copper (weight %)	0.27	0.05	0.31
Nickel (weight %)	0.59	0.96	0.59
Chemistry Factor	183	68	196.7
End of Life Inside Diameter Fluence (10^{19} n/cm^2)	0.041	0.041	0.19
ΔRT_{NDT} (°F)	48	18	109.4
$\Delta RT_{NDT(U)}$ (°F)	-5	-32	20
Mean RT_{NDT} (°F)	43	-14	129.4

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Generic Letter 98-05, Criterion 2

Demonstrate that licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the NRC's July 28, 1998, safety evaluation.

Response

EGC has procedures in place for DNPS, Units 2 and 3, and QCNPS, Units 1 and 2, that guide operators in controlling and monitoring reactor pressure during all phases of operation, including cold shutdown. Use of these procedures minimizes the potential for low temperature over-pressurization (LTOP) events, and is reinforced through operator training. A Primary System Leakage test is performed prior to each restart after a refueling outage. The associated station test procedure has sufficient guidance to minimize the likelihood of an LTOP event, and requires a pre-job briefing prior to test commencement with all involved personnel. During pressure testing, measures are taken to limit the potential for system perturbations that could lead to pressure transients. These measures include both administrative and/or hardware controls, such as limiting testing or work activities, or installing jumpers to defeat system actuations that are not required operable. RPV temperature and pressure are required to be monitored and controlled to within the Technical Specifications pressure and temperature (P/T) limits curve during all portions of the testing. The normal and contingency methods to enact pressure control are specified in the test procedure.

A designated Test Coordinator is responsible for the coordination of the test (i.e., from initiation to conclusion) and maintains cognizance of test status. A controlled rate of pressure increase is administratively limited in the test procedure to no greater than 30 pounds per square inch (psi) per minute at DNPS, and not greater than 50 psi per minute at QCNPS. If the rate of pressurization exceeds this limit, a contingency sequence portion of the testing procedures provides directions to reduce the rate of pressure increase by depressurizing through the Reactor Water Cleanup System, securing Control Rod Drive (CRD) pumps, and opening the main steam drain lines.

Other than the CRD system, the other high pressure coolant sources that could inadvertently initiate and result in an LTOP event are the Condensate/Feedwater, the Safe Shutdown Makeup Pump (SSMP) at QCNPS, Reactor Core Isolation Cooling (RCIC) at QCNPS, and High Pressure Coolant Injection (HPCI) Systems.

During a normal RPV fill sequence prior to pressure testing, the Condensate System is used to fill the reactor. This evolution is carefully controlled per the test procedure to minimize the potential for an LTOP. The feedwater pump motors are prevented from starting by the reactor water level high feedwater pump trip signal, which is present due

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to the high reactor water levels required during pressure testing. The SSMP is a manually operated system that has no automatic initiation signals. Initiation of the SSMP is strictly governed by station procedures. During pressure testing, the reactor is in cold shutdown, and as a result, there is no steam available to drive the turbine driven RCIC or HPCI pumps. In addition, the HPCI and RCIC steam supply and pump discharge valves are closed and their associated motor operator breakers are opened in accordance with the test procedures.

The Standby Liquid Control (SLC) system is also a high pressure water source to the RPV. Similar to the SSMP, there are no automatic initiation signals associated with this system. Operation of the SLC system is strictly governed by station emergency operating procedures, and requires an operator to manually start the system from the main control room via a keylock switch manipulation.

The low pressure coolant sources include the Emergency Core Cooling Systems (ECCS) (i.e., Core Spray and Residual Heat Removal) and the Condensate System. Operation of the ECCS systems is also governed by station emergency operating procedures. Although certain automatic initiation signals are required operable during pressure testing, an ECCS actuation would occur only when reactor conditions warranted RPV injection (for example, during a low water level condition). In addition, the shutoff head of the ECCS pumps is relatively low and the injection valves are interlocked closed at pressures greater than approximately 300 psig. For these reasons, an LTOP event that would exceed the P/T curve limits due to an inadvertent ECCS injection is considered unlikely. As mentioned above, the Condensate System is normally used for RPV fill and is carefully governed by the test procedure.

During cold shutdown when the reactor head is tensioned, an LTOP event is prevented by the normal unit shutdown procedure, which requires the operator to place the RPV head vent valves in an open position when reactor coolant temperatures are below 190°F.

In addition to the procedural barriers, licensed operators are provided specific training on the P/T curves and requirements of the Technical Specifications. Simulator sessions are conducted which include plant heat-up and cool-down. Additionally, in response to industry operating experience, the operating training program is routinely evaluated and revised, as necessary, to reduce the possibility of events such as an LTOP.

Summary

In summary, EGC has reviewed the methodology used in Reference 2, and considering DNPS and QCNPS plant specific materials properties, fluence, operational practices, and the provisions of Reference 3, the criteria established in Generic Letter 98-05 (i.e., Reference 4) are satisfied. Therefore, permanent relief is requested from the examination requirements of the ASME Code, Section XI, Subarticle IWB-2500, Table IWB-2500-1, Examination Category B-A, Item Number B1.11 for RPV circumferential shell welds since the proposed alternative provides an acceptable level of quality and safety in accordance with 10CFR50.55a(a)(3)(i).

6. Duration of Proposed Alternative:

Permanent relief is requested for the additional license operating period requested in Reference 1 for DNPS, Units 2 and 3, and QCNPS, Units 1 and 2. Although Reference 4 permits BWR licensees to request permanent relief for the remaining term of the existing initial operating license, EGC has demonstrated that the criteria specified in Reference 4 will continue to be met for the entire additional operating period requested in Reference 1. Therefore, the requested duration of the proposed alternative is justified.

7. Precedents:

The NRC has previously approved similar relief for several nuclear power plants, including DNPS Units 2 and 3 (i.e., Docket Numbers 50-237 and 50-249, TAC Nos. MA6228 and MA6229), and Susquehanna Steam Electric Station Units 1 and 2 (i.e., Docket Numbers 50-387 and 50-388, TAC Nos. MB0484 and MB0485). The relief request for DNPS Units 2 and 3 was submitted to the NRC in Reference 10, and the NRC granted the relief in Reference 11. The relief request for Susquehanna Steam Electric Station Units 1 and 2 was submitted to the NRC in Reference 12, and the NRC granted the relief in Reference 13.

8. References:

1. Letter from J. A. Benjamin (Exelon Generation Company, LLC) to U. S. NRC, “Application for Renewed Operating Licenses,” dated January 3, 2003

2. BWRVIP-05, “BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05),” dated September 28, 1995

3. Letter from G. C. Lainas (U. S. Nuclear Regulatory Commission) to C. Terry (BWRVIP), “Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925),” dated July 28, 1998

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4. NRC Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998

5. NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," dated August 2000

6. Letter from S. A. Richards (U. S. NRC) to J. F. Klapproth (GE Nuclear Energy), "Safety Evaluation for NEDC-32983P, 'General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation' (TAC No. MA9891)," dated September 14, 2001

7. NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001

8. Letter from L. W. Rossbach (U. S. NRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Dresden Nuclear Power Station, Units 2 and 3 – Issuance of Amendments for Extended Power Uprate (TAC Nos. MB0844 and MB0845)," dated December 21, 2001

9. Letter from S. N. Bailey (U. S. NRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Quad Cities Nuclear Power Station, Units 1 and 2 – Issuance of Amendments for Extended Power Uprate (TAC Nos. MB0842 and MB0843)," dated December 21, 2001

10. Letter from J. M. Heffley (Commonwealth Edison Company) to U. S. Nuclear Regulatory Commission, "Relief Request for Alternative Weld Examination of Circumferential Reactor Pressure Vessel Shell Welds," dated July 26, 1999

11. Letter from A. J. Mendiola (U. S. Nuclear Regulatory Commission) to O. D. Kingsley (Commonwealth Edison Company), "Dresden – Authorization for Proposed Alternative Reactor Pressure Vessel Circumferential Weld Examinations (TAC Nos. MA6228 and MA6229)," dated February 25, 2000

12. Letter from R. G. Byram (PPL Susquehanna, LLC) to U. S. Nuclear Regulatory Commission, "Request for Alternative to 10CFR50.55a Examination Requirements of Examination Category B1.11 Reactor Pressure Vessel Welds for PPL Susquehanna LLC Units 1 and 2 PLA-5251," dated November 7, 2000

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13. Letter from M. Gamberoni (U. S. Nuclear Regulatory Commission) to R. G. Byram (PPL Susquehanna, LLC), "Relief Request No. 22 (RR-22) from American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Susquehanna Steam Electric Station Units 1 and 2 (TAC Nos. MB0484 and MB0485)," dated February 28, 2001

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Request for Relief for the Use of BWRVIP Guidelines in Lieu of Specific ASME Code Requirements on Reactor Pressure Vessel Internals and Components Inspection In Accordance with 10CFR50.55a(a)(3)(i) Alternative Provides Acceptable Level of Quality and Safety

1. ASME Code Component(s) Affected:

Code Class:	1
Reference:	IWB-2500, Table IWB-2500-1
Examination Category:	B-N-1, B-N-2
Item Number:	B13.10, B13.20, B13.30, and B13.40
Description:	Use of BWRVIP Guidelines in Lieu of Specific ASME Code Requirements on Reactor Pressure Vessel Internals and Components Inspection
Component Numbers:	Vessel Interior, Interior Attachments within Beltline Region, Interior Attachments beyond Beltline Region, and Core Support Structure

2. Applicable Code Edition and Addenda:

The Inservice Inspection Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2007 Edition through the 2008 Addenda.

3. Applicable Code Requirements:

ASME Section XI requires the examination of components within the Reactor Pressure Vessel. These examinations are included in Table IWB-2500-1, Examination Categories B-N-1 and B-N-2, and identified with the following item numbers:

B13.10 Examine accessible areas of the reactor vessel interior each period by the VT-3 method (B-N-1).

B13.20 Examine interior attachment welds within the beltline region each interval by the VT-1 method (B-N-2).

B13.30 Examine interior attachment welds beyond the beltline region each interval by the VT-3 method (B-N-2).

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B13.40	Examine surfaces of the welded core support structure each interval by the VT-3 method (B-N-2).

These examinations are performed to assess the structural integrity of components within the boiling water reactor pressure vessel.

4. Reason for Request:

Pursuant to 10CFR50.55a(a)(3)(i), relief is requested for the proposed alternative to the Code requirements provided above on the basis that the use of the BWRVIP guidelines discussed below will provide an acceptable level of quality and safety.

The BWRVIP Inspection and Evaluation (I&E) guidelines have recommended aggressive specific inspection by BWR operators to completely identify material condition issues with BWR components. A wealth of inspection data has been gathered during these inspections across the BWR industry. I&E guidelines focus on specific and susceptible components, specify appropriate inspection methods capable of identifying real anticipated degradation mechanisms, and require re-examination at conservative intervals. In contrast, the code inspection requirements were prepared before the BWRVIP initiative and have not evolved with BWR inspection experience.

Use of this proposed alternative will maintain an adequate level of quality and safety and avoid unnecessary inspections.

5. Proposed Alternative and Basis for Use:

In lieu of the requirements of ASME Section XI, the proposed alternative is detailed in attached Table 1 for Examination Category B-N-1 and B-N-2.

Dresden Nuclear Power Station Units 2 and 3 will satisfy the Examination Category B-N-1 and B-N-2 requirements as described in Table 1 in accordance with BWRVIP guideline requirements. This relief request proposes to utilize the identified BWRVIP guidelines in lieu of the associated Code requirements, including examination method, examination volume, frequency, training, successive and additional examinations, flaw evaluations, and reporting.

Not all the components addressed by these guidelines are code components. The following guidelines are applicable to this Relief Request:

- BWRVIP-03, "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines"

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- BWRVIP-18, Rev. 1, "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines"
- BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines"
- BWRVIP-26-A, "BWR Top Guide Inspection and Flaw Evaluation Guidelines"
- BWRVIP-27-A, "BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines"
- BWRVIP-38, "BWR Shroud Support Inspection and Flaw Evaluation Guidelines"
- BWRVIP-47-A, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"
- BWRVIP-48-A, "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines"
- BWRVIP-76, Rev. 1, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines"
- BWRVIP-94, Rev. 2, "BWR Vessel and Internals Project Program Implementation Guide"
- BWRVIP-138, Rev. 1, "Updated Jet Pump Beam Inspection and Flaw Evaluation"
- BWRVIP-183, "BWR Vessel and Internals Project, Top Guide Grid Beam Inspection and Flaw Evaluation Guidelines"

Inspection services, by an Authorized Inspection Agency, will be applied to the proposed alternative actions of this relief request.

BWRs now examine reactor internals in accordance with BWRVIP guidelines. These guidelines have been written to address the safety significant vessel internal components and to examine and evaluate the examination results for these components using appropriate methods and reexamination frequencies. The BWRVIP has established a reporting protocol for examination results and deviations. The NRC has agreed with the BWRVIP approach as documented in References 1 through 10. Therefore, use of these guidelines, as an alternative to the subject Code requirements, provides an acceptable level of quality and safety and will not adversely impact the health and safety of the public.

As additional justification, Attachment 1 ("Comparison of Code Examination Requirements to BWRVIP Examination Requirements") provides specific examples which compare the inspection requirements of ASME Code Item Numbers B13.10, B13.20, B13.30, and B13.40 in Table IWB-2500-1, to the inspection requirements in the BWRVIP documents. Specific BWRVIP documents are provided as examples. This comparison also includes a discussion of the inspection methods. These comparisons demonstrate that use of these guidelines, as an alternative to the subject Code requirements, provides an acceptable level of quality and safety and will not adversely impact the health and safety of the public.

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From outage D2R19 (Unit 2) and outage D3R19 (Unit 3), and since the conditional authorization of the Fourth Interval BWRVIP Relief Request, Revision 1 on April 30, 2008, no indications or flaws have been found on the BWRVIP subject welds in subsequent examinations.

Table 1 compares present ASME Examination Category B-N-1 and B-N-2 requirements with the above current BWRVIP guideline requirements, as applicable, to Dresden Nuclear Power Station. Therefore, Table 1 only represents a current comparison. Any deviations from the BWRVIP guidelines referenced within this relief request for the duration of the proposed alternative will be appropriately documented and communicated to the NRC, per the BWRVIP Deviation Disposition Process.

Also, the reactor vessel internals inspection program at Dresden Nuclear Power Station has been developed and implemented to satisfy the requirements of BWRVIP-94. It is recognized that the BWRVIP executive committee periodically revises the BWRVIP guidelines to include enhancements in inspection techniques and flaw evaluation methodologies. Where the revised version of a BWRVIP inspection guideline continues to also meet the requirements of the version of the BWRVIP inspection guideline that forms the safety basis for an NRC-authorized proposed alternative to the requirements of 10CFR50.55a, it may be implemented. Otherwise, the revised guidelines will only be implemented after NRC approval of the revised BWRVIP guidelines or a plant-specific request for relief has been approved.

6. <u>Duration Of Proposed Alternative:</u>

Relief is requested for the Fifth Ten-Year Inspection Interval for Dresden Nuclear Power Station Units 2 and 3.

7. <u>Precedents:</u>

Exelon/Amergen BWR fleet as discussed in Reference 11. The Exelon/Amergen Fleetwide Relief Request for BWRVIP was authorized conditionally per SE dated April 30, 2008. The Fifth Inspection Interval Relief Request utilizes a similar approach that was previously approved.

Perry Nuclear Power Plant, Unit No. 1 as discussed in Reference 12. Perry Nuclear Power Plant, Unit No. 1 was authorized per NRC SE dated January 31, 2012.

Fermi 2 as discussed in Reference 13. Fermi 2 was authorized per NRC SE dated February 17, 2012.

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8. References:

1. Letter NRC to BWRVIP, "Final Safety Evaluation for Electric Power Research Institute Boiling Water Reactor Vessel and Internals Project Technical Report 1016568, 'BWRVIP-18, Revision 1: BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (TAC No. ME2189),'" dated January 30, 2012 (ML113620684)

2. Letter NRC to BWRVIP, "NRC Approval Letter of BWRVIP-26-A, 'BWR Vessel and Internals Project Boiling Water Reactor Top Guide Inspection and Flaw Evaluation Guidelines,'" dated September 9, 2005

3. Letter NRC to BWRVIP, Proprietary Version of NRC Staff Review of BWRVIP-27-A, "BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines," dated June 10, 2004

4. Letter NRC to BWRVIP, "Final Safety Evaluation of the 'BWR Vessel and Internals Project, BWR Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38),' EPRI Report TR-108823 (TAC No. M99638)," dated July 24, 2000

5. Letter NRC to BWRVIP, "NRC Approval Letter of BWRVIP-47-A, 'BWR Vessel and Internals Project Boiling Water Reactor Lower Plenum Inspection and Flaw Evaluation Guidelines,'" dated September 9, 2005

6. Letter NRC to BWRVIP, "NRC Approval Letter of BWRVIP-48-A, 'BWR Vessel and Internals Project Vessel ID Attachment Weld Inspection and Flaw Evaluation Guideline,'" dated July 25, 2005

7. "BWRVIP-76NP, Rev. 1: BWR Vessel and Internals Project BWR Core Shroud Inspection and Flaw Evaluation Guidelines," dated May 2011 (ML11195A182)

8. Letter from Chairman, BWR Vessel and Internals Project to NRC, "Project No. 704 - BWRVIP Program Implementation Guide (BWRVIP-94NP, Revision 2)," dated September 22, 2011 (ML11271A058)

9. "BWRVIP-138NP, Revision 1: BWR Vessel and Internals Project, Updated Jet Pump Beam Inspection and Flaw Evaluation Guidelines," dated January 2009 (ML090760986)

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10. "BWRVIP-183NP: BWR Vessel and Internals Project Top Guide Grid Beam Inspection and Flaw Evaluation Guidelines," dated December 7, 2007 (ML080220433)

11. Letter from NRC to Exelon/Amergen, "Clinton Power Station, Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; LaSalle County Station, Units 1 and 2; Limerick Generating Station, Units 1 and 2; Oyster Creek Nuclear Generating Station; Peach Bottom Atomic Power Station, Units 2 and 3; and Quad Cities Nuclear Power Station, Units 1 and 2 – Relief Request to Use Boiling Water Reactor Vessel and Internals Project Guidelines in Lieu of Specific ASME Code Requirements (TAC Nos. MD5352 through MD5363)," dated April 30, 2008 (ML080980311)

12. Letter from NRC to FirstEnergy Nuclear Operating Company (Perry Nuclear Power Plant, Unit No. 1), "Perry Nuclear Power Plant, Unit No. 1, RE: Safety Evaluation in Support of 10CFR50.55a Requests for the Third 10-Year In-Service Inspection Interval (TAC Nos. ME5373, ME5376, ME5377, ME5379, and ME5380)," dated January 31, 2012 (ML120180372)

13. Letter from NRC to Detroit Edison Company (Fermi 2), "Fermi 2- Evaluation of Applicable 10-Year Interval Inservice Inspection Relief Request – Use of Boiling Water Reactor Vessel and Internals Project (BWRVIP) Guidelines in Lieu of Specific ASME Code Requirements (TAC No. ME6765)," dated February 17, 2012 (ML120370286)

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TABLE 1
Comparison of ASME Examination Category B-N-1 and B-N-2 Requirements With BWRVIP Guidance Requirements[1]

ASME Item Number, Table IWB-2500-1	Component	ASME Exam Scope	ASME Exam	ASME Frequency	Authorized Alternative	BWRVIP Exam Scope	BWRVIP Exam	BWRVIP Frequency
B13.10 Reactor Vessel Interior		Accessible Areas	VT-3	Each period	BWRVIP-18-R1, 25, 26-A, 27-A, 38, 47-A, 48-A, 76-R1, and 138-R1	Overview examinations of components during BWRVIP examinations satisfy Code VT-3 visual inspection requirements.		
B13.20 Interior Attachments Within Beltline Region	Jet Pump Riser Braces	Accessible Welds	VT-1	Each 10-year Interval	BWRVIP-48-A, Table 3-2	Riser Brace Attachment	EVT-1	100% in first 12 years (with 50% to be inspected in the first 6 years); 25% during each subsequent 6 years
	Lower Surveillance Specimen Holder Brackets				BWRVIP-48-A, Table 3-2	Bracket Attachment	VT-1	Each 10-year Interval
B13.30 Interior Attachments Beyond Beltline	Guide Rod Brackets	Accessible Welds	VT-3	Each 10-year Interval	BWRVIP-48-A, Table 3-2	Bracket Attachment	VT-3	Each 10-year Interval
	Steam Dryer Support Brackets				BWRVIP-48-A, Table 3-2	Bracket Attachment	EVT-1	Each 10-year Interval
	Feedwater Sparger Brackets				BWRVIP-48-A, Table 3-2	Bracket Attachment	EVT-1	Each 10-year Interval
	Core Spray Piping Brackets				BWRVIP-48-A, Table 3-2	Bracket Attachment	EVT-1	Every 4 Refueling Cycles
	Upper Surveillance Specimen Holder Brackets				BWRVIP-48-A, Table 3-2	Bracket Attachment	VT-3	Each 10-year Interval

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TABLE 1
Comparison of ASME Examination Category B-N-1 and B-N-2 Requirements With BWRVIP Guidance Requirements[1]

ASME Item Number, Table IWB-2500-1	Component	ASME Exam Scope	ASME Exam	ASME Frequency	Authorized Alternative	BWRVIP Exam Scope	BWRVIP Exam	BWRVIP Frequency
	Shroud Support (Weld H9)	(Rarely Accessible)			BWRVIP-38, 3.1.3.2, Figures 3-2 and 3-5	Weld H9[2]	EVT-1 or UT	Maximum of 6 years for one-sided EVT-1, Maximum of 10 years for UT
	Shroud Support Legs (Weld H12)				BWRVIP-38, 3.2.3	Weld H12	Per BWRVIP-38 NRC SE (7-24-2000), inspect with appropriate method[4]	When accessible
B13.40 Welded Core Support Structure	Shroud Support (Weld H10)	Accessible Surfaces	VT-3	Each 10-year Interval	BWRVIP-38, 3.1.3.2, Figure 3-2 and 3-5	Shroud Support (Weld H10) and Leg Welds	EVT-1 or UT	Based on as found conditions, to a maximum 6 years for one-sided EVT-1, 10 years for UT where accessible
	Shroud Vertical Welds				BWRVIP-76-R1, 3.3, Figure 3-1 and 3-3	Vertical and Ring Segment Welds as applicable	EVT-1 or UT	Maximum 6 years for one-sided EVT-1, 10 years for UT

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TABLE 1
Comparison of ASME Examination Category B-N-1 and B-N-2 Requirements With BWRVIP Guidance Requirements(1)

ASME Item Number, Table IWB-2500-1	Component	ASME Exam Scope	ASME Exam	ASME Frequency	Authorized Alternative	BWRVIP Exam Scope	BWRVIP Exam	BWRVIP Frequency
	Shroud Repairs(3)				BWRVIP-76-R1, Section 3.5	Tie-Rod Repair	VT-3	Per designer recommendations per BWRVIP-76-R1

NOTES:
1) This Table provides only an overview of the requirements. For more details, refer to ASME Section XI, Table IWB-2500-1, and the appropriate BWRVIP document.
2) In accordance with Appendix A of BWRVIP-38, a site specific evaluation will determine the minimum required weld length to be examined.
3) Shroud repairs are currently installed on both units at Dresden Nuclear Power Station.
4) When inspection tooling and methodologies are available, they will be utilized to establish a baseline inspection of these welds.

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ATTACHMENT 1
Comparison of Code Examination Requirements to BWRVIP Examination Requirements

The following discussion provides a comparison of the examination requirements provided in ASME Code Item Numbers B13.10, B13.20, B13.30, and B13.40 in Table IWB-2500-1, to the examination requirements in the BWRVIP guidelines. Specific BWRVIP guidelines are provided as examples for comparisons. This comparison also includes a discussion of the examination methods.

1. **Code Requirement - B13.10 - Reactor Vessel Interior Accessible Areas (B-N-1)**

The ASME Section XI Code requires a VT-3 visual examination of reactor vessel accessible areas, which are defined as the spaces above and below the core made accessible during normal refueling outages. The frequency of these examinations is specified as the first refueling outage, and at intervals of approximately 3 years, during the First Inspection Interval, and each period during each successive 10-year Inspection Interval. Typically, these examinations are performed every other refueling outage of the Inspection Interval. This examination requirement is a non-specific requirement that is a departure from the traditional ASME Section XI examinations of welds and surfaces. As such, this requirement has been interpreted and satisfied differently across the domestic fleet. The purpose of the examination is to identify relevant conditions such as distortion or displacement of parts; loose, missing, or fractured fasteners; foreign material, corrosion, erosion, or accumulation of corrosion products; wear; and structural degradation.

Portions of the various examinations required by the applicable BWRVIP guidelines require access to accessible areas of the reactor vessel during each refueling outage. Examination of core spray piping and spargers (BWRVIP-18-R1), top guide (BWRVIP-26-A), jet pump welds and components (BWRVIP-138-R1), interior attachments (BWRVIP-48-A), core shroud welds (BWRVIP-76-R1), shroud support (BWRVIP-38), and lower plenum components (BWRVIP-47-A) provides such access. Locating and examining specific welds and components within the reactor vessel areas above, below (if accessible), and surrounding the core (annulus area) entails access by remote camera systems that essentially perform equivalent VT-3 visual examination of these areas or spaces as the specific weld or component examinations are performed. This provides an equivalent method of visual examination on a more frequent basis than that required by the ASME Section XI Code. Evidence of wear, structural degradation, loose, missing, or displaced parts, foreign materials, and corrosion product buildup can be, and has been observed during the course of implementing these BWRVIP examination requirements. Therefore, the specified BWRVIP Guideline requirements meet or exceed the subject Code requirements for examination method and frequency of

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ATTACHMENT 1
Comparison of Code Examination Requirements to BWRVIP Examination Requirements

the interior of the reactor vessel. Accordingly, these BWRVIP examination requirements provide an acceptable level of quality and safety as compared to the subject Code requirements.

2. **Code Requirement - B13.20 - Interior Attachments Within the Beltline (B-N-2)**

The ASME Section XI Code requires a VT-1 visual examination of accessible reactor interior surface attachment welds within the beltline each 10-year interval. In the model 3 boiling water reactor, this includes the jet pump riser brace welds-to-vessel wall and the lower surveillance specimen support bracket welds-to-vessel wall. In comparison, the BWRVIP requires the same examination method and frequency for the lower surveillance specimen support bracket welds, and requires an EVT-1 visual examination on the remaining attachment welds in the beltline region in the first 12 years, and then 25% during each subsequent 6 years.

The jet pump riser brace examination requirements are provided below to show a comparison between the Code and the BWRVIP examination requirements.

Comparison to BWRVIP Requirements - Jet Pump Riser Braces (BWRVIP-138, Rev. 1 and BWRVIP-48-A)

- The ASME Code requires a 100% VT-1 visual examination of the jet pump riser brace-to-reactor vessel wall pad welds each 10-year interval.
- The BWRVIP requires an EVT-1 visual examination of the jet pump riser brace-to-reactor vessel wall pad welds the first 12 years and then 25% during each subsequent 6 years.
- BWRVIP-48-A specifically defines the susceptible regions of the attachment that are to be examined.

The Code VT-1 visual examination is conducted to detect discontinuities and imperfections on the surfaces of components, including such conditions as cracks, wear, corrosion, or erosion. The BWRVIP enhanced VT-1 (EVT-1) visual examination is conducted to detect discontinuities and imperfections on the surface of components and is additionally specified to detect potentially very tight cracks characteristic of fatigue and inter-granular stress corrosion cracking (IGSCC), the relevant degradation mechanisms for these components. General wear, corrosion, or erosion although generally not a concern for inherently tough, corrosion resistant stainless steel material, would also be detected during the process of performing a BWRVIP EVT-1 visual examination.

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ATTACHMENT 1
Comparison of Code Examination Requirements to BWRVIP Examination Requirements

The Code VT-1 visual examination method requires that at a maximum distance of 2 feet or a letter character with a height of 0.044 inches can be read. The BWRVIP EVT-1 visual examination method requires resolution of 0.044 inch characters on the examination surface. BWRVIP-48-A includes a diagram and prescribes examination for the configuration of this plant.

The calibration standards used for BWRVIP EVT-1 visual examinations utilize the Code characters, thus assuring at least equivalent resolution compared to the ASME Code. Although the BWRVIP examination may be less frequent, it is a more comprehensive method. Therefore, the enhanced flaw detection capability of an EVT-1 visual examination, with a less frequent examination schedule provides an acceptable level of quality and safety to that provided by the ASME Code.

3. **Code Requirement - B13.30 - Interior Attachment Beyond the Beltline Region (B-N-2)**

The ASME Section XI Code requires a VT-3 visual examination of accessible reactor interior surface attachment welds beyond the beltline each 10-year interval. In the BWR/3 model, this includes the core spray piping primary and supplemental support bracket welds-to-vessel wall, the upper surveillance specimen support bracket welds-to-vessel wall, the feedwater sparger support bracket welds-to-reactor vessel wall, the steam dryer support welds-to-reactor vessel wall, the guide rod support bracket weld-to-reactor vessel wall, and the shroud support plate-to-vessel weld. BWRVIP-48-A requires as a minimum the same VT-3 visual examination method as the Code for some of the interior attachment welds beyond the beltline region, and in some cases specifies an enhanced visual examination technique EVT-1 visual examination for these welds. For those interior attachment welds that have the same VT-3 method of visual examination, the same scope of examination (accessible welds), the same examination frequency (each 10-year interval) and ASME Section XI flaw evaluation criteria, the level of quality and safety provided by the BWRVIP requirements are equivalent to that provided by the ASME Code.

For the core spray primary and secondary support bracket attachment welds, the steam dryer support bracket attachment welds, the feedwater sparger support bracket attachment welds, and the shroud support plate-to-vessel welds, as applicable, the BWRVIP guidelines require an EVT-1 visual examination at the same frequency as the Code. Therefore, the BWRVIP requirements provide the same level of quality and safety to that provided by the ASME Code.

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ATTACHMENT 1
Comparison of Code Examination Requirements to
BWRVIP Examination Requirements

The core spray piping bracket-to-vessel attachment weld is used as an example for comparison between the Code and BWRVIP examination requirements as discussed below.

Comparison to BWRVIP Requirements - Core Spray piping Bracket Welds (BWRVIP-48-A)

- The Code examination requirement is a VT-3 visual examination of each weld every 10 years.
- The BWRVIP examination requirement is an EVT-1 visual examination for the Core Spray piping bracket attachment welds with each weld examined every four cycles (8 years for units with a two year fuel cycle).

The BWRVIP examination method EVT-1 visual examination has superior flaw detection and sizing capability, and the same flaw evaluation criteria are used.

The Code VT-3 visual examination is conducted to detect component structural integrity by ensuring the components general condition is acceptable. An enhanced EVT-1 visual examination is conducted to detect discontinuities and imperfections on the examination surfaces, including such conditions as tight cracks caused by IGSCC or fatigue, the relevant degradation mechanisms for BWR internal attachments.

Therefore, with the EVT-1 visual examination method, the same examination scope (accessible welds), the same examination frequency, the same flaw evaluation criteria (ASME Section XI), the level of quality and safety required by the BWRVIP criteria is superior than that required by the Code.

4. Code Requirement - B13.40 - Welded Core Support Structures (B-N-2)

The ASME Code requires a VT-3 visual examination of accessible surfaces of the welded core support structure each 10-year interval. In the boiling water reactor, the welded core support structure has primarily been considered the shroud. In later designs, the shroud itself is considered part of the welded core support structure. Historically, this requirement has been interpreted and satisfied differently across the industry. The proposed alternate examination replaces this ASME requirement with specific BWRVIP guidelines that examine susceptible locations for known relevant degradation mechanisms.

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ATTACHMENT 1
Comparison of Code Examination Requirements to BWRVIP Examination Requirements

Comparison to BWRVIP Requirements - Shroud Supports (BWRVIP-38)

- The Code requires a VT-3 visual examination of accessible surfaces each 10-year interval.
- The BWRVIP requires either an enhanced visual examination technique (EVT-1) every 6 years or volumetric examination (UT) every 10 years as compared to the Code requirement (VT-3). (Only 10% of the weld is required to be examined.)

BWRVIP recommended examinations of welded core support structures are focused on the known susceptible areas of this structure, including the welds and associated weld heat affected zones. In many locations, the BWRVIP guidelines require a volumetric examination of the susceptible welds at a frequency identical to the Code requirement.

Shroud repair tie-rods have been installed at Dresden Nuclear Power Station; therefore, the BWRVIP referenced examinations are the same as the Code requirements. Shroud repair tie-rod examinations are recommended in BWRVIP-76-R1, and have the same basic VT-3 method of visual examination, the same scope of examination (accessible surfaces), the same examination frequency (each 10-year interval) and the same flaw evaluation criteria. Therefore, the BWRVIP requirements provide a level of quality and safety equivalent to that provided by the ASME Code.

For other welded core support structure components, the BWRVIP requires an EVT-1 or UT of core support structures. The core shroud is used as an example for comparison between the Code and BWRVIP examination requirements as shown below.

Comparison to BWRVIP Requirements - BWR Core Shroud Examination and Flaw Evaluation Guideline (BWRVIP-76-A, Rev. 1)

- The Code requires a VT-3 visual examination of accessible surfaces each 10-year interval.
- The BWRVIP requires an EVT-1 visual examination from the inside and outside surface where accessible or ultrasonic examination of each core shroud circumferential weld that has not been structurally replaced with a shroud repair at a calculated “end of interval” (EOI) that will vary depending upon the amount of flaws present, but not to exceed ten years.

The BWRVIP recommended examinations specify locations that are known to be vulnerable to BWR relevant degradation mechanisms rather than accessible surfaces.

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ATTACHMENT 1
Comparison of Code Examination Requirements to BWRVIP Examination Requirements

The BWRVIP examination methods (EVT-1 or UT) are superior to the Code required VT-3 visual examination for flaw detection and characterization. The superior flaw detection and characterization capability and the comparable flaw evaluation criteria, results in the BWRVIP criteria providing a level of quality and safety equivalent to or superior to that required by the Code requirements.

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(RESERVED)

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Request for Relief for Use of ASME Code Case N-789, Alternative Requirements for Pad Reinforcement of Class 2 and 3 Moderate-Energy Carbon Steel Piping for Raw Water Service In Accordance with 10CFR50.55a(a)(3)(i) Alternative Provides Acceptable Level of Quality and Safety

***** NOTE 1 *****

Dresden Nuclear Power Station Units 2 and 3 Fifth Interval Relief Request I5R-09 is provided for completeness and has no technical changes from the previously approved relief request. This relief request was submitted as an Exelon Fleet request under cover letter RS-11-182 on October 7, 2011, with supplemental letters dated November 10, 2011, and February 13, 2012, and was authorized during the Fourth Interval under NRC SE dated May 10, 2012, which for Dresden Nuclear Power Station covered both the Fourth and Fifth Intervals as specified in Table 1 of the NRC SE.

No technical changes to the actual approved relief request have been made in the Fifth Interval ISI Program Plan and no further or revised authorization is required.

***** NOTE 2 *****

Per NRC SE dated May 10, 2012, the NRC staff concluded that the proposed alternative failed to meet the regulatory standard of 10CFR50.55a(a)(3)(i). However, the NRC staff further concluded that the proposed alternative provided reasonable assurance of structural integrity and leak tightness of the ASME Section XI, Class 2 and 3, moderate energy carbon steel raw water piping and that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concluded that the licensee had adequately addressed all of the regulatory requirements set forth in 10CFR50.55a(a)(3)(ii), and as such, authorized the relief request.

1. **ASME Code Component(s) Affected:**

All Class 2 and 3 moderate energy carbon steel raw water piping systems. Raw water is defined as water such as from a river, lake, or well or brackish/salt water - used in plant equipment, area coolers, and heat exchangers. In many plants it is referred to as “Service Water.” This Code Case applies to Class 2 and 3 moderate energy (i.e., less than or equal to 200°F (93°C) and less than or equal to 275 psig (1.9 MPa) maximum operating conditions) carbon steel raw water piping.

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2. Applicable Code Edition and Addenda:

PLANT	INTERVAL	EDITION	START	END
Braidwood Station, Units 1 and 2	Third	2001 Edition, through the 2003 Addenda	July 29, 2008 October 17, 2008	July 28, 2018 October 16, 2018
Byron Station, Units 1 and 2	Third	2001 Edition, through the 2003 Addenda	January 16, 2006	July 15, 2016
Clinton Power Station	Third	2004 Edition	July 1, 2010	June 30, 2020
Dresden Nuclear Power Station, Units 2 and 3	Fourth	1995 Edition, through the 1996 Addenda	January 20, 2003	January 19, 2013
Dresden Nuclear Power Station, Units 2 and 3	Fifth	2007 Edition, through the 2008 Addenda	January 20, 2013	January 19, 2023
LaSalle County Station, Units 1 and 2	Third	2001 Edition, through the 2003 Addenda	October 1, 2007	September 30, 2017
Limerick Generating Station, Units 1 and 2	Third	2001 Edition, through the 2003 Addenda	February 1, 2007	January 31, 2017
Oyster Creek Nuclear Generating Station	Fourth	1995 Edition, through the 1996 Addenda	October 15, 2002	October 14, 2012
Oyster Creek Nuclear Generating Station	Fifth	2007 Edition, through the 2008 Addenda	October 15, 2012	October 14, 2022
Peach Bottom Atomic Power Station, Units 2 and 3	Fourth	2001 Edition, through the 2003 Addenda	November 5, 2008	November 4, 2018
Quad Cities Nuclear Power Station, Units 1 and 2	Fourth	1995 Edition, through the 1996 Addenda	March 10, 2003	April 1, 2013
Quad Cities Nuclear Power Station, Units 1 and 2	Fifth	2007 Edition, through the 2008 Addenda	April 2, 2013	April 1, 2023
Three Mile Island Nuclear Station, Unit 1	Fifth	2004 Edition	April 20, 2011	April 19, 2022

3. Applicable Code Requirement:

ASME Code, Section Xl, IWA-4400 of the 1995 Edition through the 1996 Addenda, 2001 Edition through the 2003 Addenda, 2004 Edition, and 2007 Edition through the 2008 Addenda provides requirements for welding, brazing, metal removal, and installation of repair/replacement activities.

4. Reason for Request:

In accordance with 10CFR50.55a(a)(3)(i), Exelon Generation Company, LLC (Exelon) is requesting a proposed alternative from the requirement for replacement or internal weld repair of wall thinning conditions resulting from degradation in Class 2 and 3 moderate energy carbon steel raw water piping systems in accordance with IWA-4000. Such degradation may be the result of mechanisms such as erosion, corrosion, cavitation, and

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pitting – but excluded are conditions involving flow-accelerated corrosion (FAC), corrosion-assisted cracking, or any other form of cracking. IWA-4000 requires repair or replacement in accordance with the Owner's Requirements and the original or later Construction Code. Other alternative repair or evaluation methods are not always practicable because of wall thinness and/or moisture issues.

The primary reason for this request is to permit installation of a technically sound temporary repair to provide adequate time for evaluation, design, material procurement, planning and scheduling of appropriate permanent repair or replacement of the defective piping, considering the impact on system availability, maintenance rule applicability, and availability of replacement materials.

5. Proposed Alternative and Basis for Use:

In accordance with 10CFR50.55a(a)(3)(i), Exelon proposes to implement the requirements of ASME Code Case N-789 ("Alternative Requirements for Pad Reinforcement of Class 2 and 3 Moderate-Energy Carbon Steel Piping for Raw Water Service, Section XI, Division 1") as a temporary repair of degradation in Class 2 and 3 moderate energy raw water piping systems resulting from mechanisms such as erosion, corrosion, cavitation, or pitting, but excluding conditions involving flow-accelerated corrosion (FAC), corrosion-assisted cracking, or any other form of cracking. These types of defects are typically identified by small leaks in the piping system or by pre-emptive non-code required examinations performed to monitor the degradation mechanisms.

ASME Code Case N-789, which is included as part of this relief request, is attached. (*Note: ASME Code Case N-789 is not attached to this document*)

The alternative repair technique described in ASME Code Case N-789 involves the application of a metal reinforcing pad welded to the exterior of the piping system, which reinforces the weakened area and restores pressure integrity. This repair technique will be utilized when it is determined that this temporary repair method is suitable for the particular defect or degradation being resolved.

The Code Case requires that the cause of the degradation be determined, and that the extent and rate of degradation in the piping be evaluated to ensure that there are no other unacceptable locations within the surrounding area that could affect the integrity of the repaired piping. The area of evaluation will be dependent on the degradation mechanism present. A baseline thickness examination will be performed for a completed structural pad, attachment welds, and surrounding area, followed by monthly thickness monitoring for the first three months, with subsequent frequency based on the results of this monitoring, but at a minimum of quarterly. Areas containing pressure pads shall be visually observed at least once per month to monitor for evidence of leakage. If the areas

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containing pressure pads are not accessible for direct observation, then monitoring will be accomplished by visual assessment of surrounding areas or ground surface areas above pressure pads on buried piping, or monitoring of leakage collection systems, if available.

The repair will be considered to have a maximum service life of the time until the next refueling outage, when a permanent repair or replacement must be performed. Additional requirements for design of reinforcement pads, installation, examination, pressure testing, and inservice monitoring are provided in ASME Code Case N-789.

Based on the above justification, the use of ASME Code Case N-789 as a proposed alternative to the requirements of ASME Section XI will provide an acceptable level of quality and safety.

All other ASME Section XI requirements for which relief was not specifically requested and authorized by the NRC staff will remain applicable including third party review by the Authorized Nuclear Inservice Inspector.

ASME Code Case N-789 was approved by the ASME Board on Nuclear Codes and Standards on June 25, 2011; however, it has not been incorporated into NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," and thus is not available for application at nuclear power plants without specific NRC approval. Therefore, Exelon requests use of this alternative repair technique described in the Code Case via this relief request.

6. Duration of Proposed Alternative:

The proposed alternative is for use of the Code Case for the remainder of each plant's ten (10) year inspection interval as specified in Section 2.

Any reinforcing pads installed before the end of the ten-year inservice inspection interval will be removed during the next refueling outage, even if that refueling outage occurs after the end of the ten-year interval.

7. Precedents:

A similar repair relief request (RR-3-43) was approved for Indian Point Nuclear Generating Unit No. 3 per NRC SE dated February 22, 2008.

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Request for Relief for Expanded Applicability for Use of ASME Code Case N-532-4, Repair/Replacement Activity Documentation Requirements and Inservice Summary Report Preparation and Submission In Accordance with 10CFR50.55a(a)(3)(i) Alternative Provides Acceptable Level of Quality and Safety

1. ASME Code Components Affected:

Code Class:	1, 2, 3, and MC
Reference:	IWA-2441(b), ASME Code Case N-532-4
Examination Category:	NA
Item Number:	NA
Description:	Expanded Applicability for Use of ASME Code Case N-532-4, "Repair/Replacement Activity Documentation Requirements and Inservice Summary Report Preparation and Submission"
Component Number:	NA

2. Applicable Code Edition and Addenda:

The Inservice Inspection program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2007 Edition through the 2008 Addenda.

3. Applicable Code Requirement:

IWA-2441(b) requires Code Cases be applicable to the Edition and Addenda specified in the Inspection Plan.

ASME Code Case N-532-4, "Repair/Replacement Activity Documentation Requirements and Inservice Summary Report Preparation and Submission," provides requirements that may be used to document repair/replacement activities.

4. Reason for Request:

Pursuant to 10CFR50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative will provide an acceptable level of quality and safety.

On January 20, 2013, Dresden Nuclear Power Station Units 2 and 3 will start its Fifth Ten-Year Interval ISI Program under the requirements of the 2007 Edition through the

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2008 Addenda of ASME Section XI. When implementing this edition of ASME Section XI, Paragraph IWA-2441(b) requires code cases be applicable to the Edition and Addenda specified in the Inspection Plan.

ASME Code Case N-532-4 has an applicability limited up to the 2004 Edition through the 2005 Addenda, which is identified in the latest applicability index for ASME Section XI Code Cases. Since ASME Code Case N-532-4 only applies up to the 2004 Edition through the 2005 Addenda, Paragraph IWA-2441(b) does not allow the use of ASME Code Case N-532-4 for the Dresden Nuclear Power Station Fifth Ten-Year Interval ISI Program.

5. Proposed Alternative and Basis for Use:

Dresden Nuclear Power Station requests the applicability of ASME Code Case N-532-4 be extended to the 2007 Edition through the 2008 Addenda for use in the plant's Fifth Interval ISI Program. The NRC has accepted the use of ASME Code Case N-532-4 as an acceptable method for repair/replacement activity documentation requirements and inservice summary report preparation and submission in the latest revision of Regulatory Guide 1.147, Revision 16.

No technical changes to ASME Code Case N-532-4 are being proposed in this relief request. This relief request is being submitted to correct a timing situation, which has resulted from the application of the 2007 Edition through the 2008 Addenda of ASME Section XI for Dresden Nuclear Power Station. Since no technical change is proposed in this relief request, Dresden Nuclear Power Station considers that this alternative provides an acceptable level of quality and safety, and is consistent with provisions of 10CFR50.55a(a)(3)(i).

6. Duration of Proposed Alternative:

Relief is requested for the Fifth Ten-Year Inspection Interval for Dresden Nuclear Power Station Units 2 and 3.

7. Precedents:

None.

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Request for Relief for Expanded Applicability for Use of ASME Code Case N-661-1, Alternative Requirements for Wall Thickness Restoration of Class 2 and 3 Carbon Steel Piping for Raw Water Service In Accordance with 10CFR50.55a(a)(3)(i) Alternative Provides Acceptable Level of Quality and Safety

1. ASME Code Components Affected:

Code Class:	2 and 3
Reference:	IWA-2441(b), ASME Code Case N-661-1
Examination Category:	NA
Item Number:	NA
Description:	Expanded Applicability for Use of ASME Code Case N-661-1, "Alternative Requirements for Wall Thickness Restoration of Class 2 and 3 Carbon Steel Piping for Raw Water Service"
Component Number:	Class 2 and 3 Piping

2. Applicable Code Edition and Addenda:

The Inservice Inspection program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2007 Edition through the 2008 Addenda.

3. Applicable Code Requirement:

IWA-2441(b) requires Code Cases be applicable to the Edition and Addenda specified in the Inspection Plan.

ASME Code Case N-661-1, "Alternative Requirements for Wall Thickness Restoration of Class 2 and 3 Carbon Steel Piping for Raw Water Service," provides requirements that may be used to restore wall thickness for raw water piping systems that have experienced internal wall thinning.

4. Reason for Request:

Pursuant to 10CFR50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative will provide an acceptable level of quality and safety.

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On January 20, 2013, Dresden Nuclear Power Station Units 2 and 3 will start its Fifth Ten-Year Interval ISI Program under the requirements of the 2007 Edition through the 2008 Addenda of ASME Section XI. When implementing this edition of ASME Section XI, Paragraph IWA-2441(b) requires code cases be applicable to the Edition and Addenda specified in the Inspection Plan.

ASME Code Case N-661-1 has an applicability limited up to the 2004 Edition through the 2005 Addenda, which is identified in the latest applicability index for ASME Section XI Code Cases. Since ASME Code Case N-661-1 only applies up to the 2004 Edition through the 2005 Addenda, Paragraph IWA-2441(b) does not allow the use of ASME Code Case N-661-1 for the Dresden Nuclear Power Station Fifth Ten-Year Interval ISI Program.

5. Proposed Alternative and Basis for Use:

Dresden Nuclear Power Station requests the applicability of ASME Code Case N-661-1 be extended to the 2007 Edition through the 2008 Addenda for use in the plant's Fifth Interval ISI Program. The NRC has accepted the use of ASME Code Case N-661-1 as an acceptable method for restoring wall thickness for raw water piping systems that have experienced internal wall thinning in the latest revision of Regulatory Guide 1.147, Revision 16.

No technical changes to ASME Code Case N-661-1 are being proposed in this relief request. This relief request is being submitted to correct a timing situation, which has resulted from the application of the 2007 Edition through the 2008 Addenda of ASME Section XI for Dresden Nuclear Power Station. Since no technical change is proposed in this relief request, Dresden Nuclear Power Station considers that this alternative provides an acceptable level of quality and safety, and is consistent with provisions of 10CFR50.55a(a)(3)(i).

6. Duration of Proposed Alternative:

Relief is requested for the Fifth Ten-Year Inspection Interval for Dresden Nuclear Power Station Units 2 and 3.

7. Precedents:

None.

9.0 REFERENCES

The references used to develop this Inservice Inspection Program Plan include:

9.1 NRC References

9.1.1 Code of Federal Regulations, Title 10, Part 50, Energy
  a. Paragraph 50.55a, "Codes and Standards"
  b. Paragraph 2, "Definitions," the definition of "Reactor Coolant Pressure Boundary"
  c. Part 50, Appendix J, Primary Reactor Containment Testing for Water Cooled Power Reactors

9.1.2 NRC Regulatory Guide 1.147, Revision 16, "Inservice Inspection Code Case Acceptability," ASME Section XI, Division 1

9.1.3 NRC Regulatory Guide 1.150, Revision 1, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examination"

9.1.4 NRC Regulatory Guide 1.192, Revision 0, "Operation and Maintenance Code Case Acceptability," ASME OM Code

9.1.5 NRC Regulatory Guide 1.193, Revision 3, "ASME Code Cases Not Approved for Use"

9.1.6 NRC Regulatory Guide 1.84, Revision 35 "Design, Fabrication, and Materials Code Case Acceptability," ASME Section III, Division 1

9.1.7 NRC Regulatory Guide 1.26, Revision 3, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste- Containing Components of Nuclear Power Plants"

9.1.8 USAS B31.1.0-1967, "Power Piping"

9.1.9 NRC Final SE related to the "Boiling Water Reactor Owner's Group (BWROG) Report, GE-NE-523-A71-0594-A, Revision 1, 'Alternate Boiling Water Reactor (BWR) Feedwater Nozzle Inspection Requirements, May 2000,' (TAC No. MA6787)," dated March 10, 2000

9.1.10 NRC Final SE related to the Boiling Water Reactor Owners' Group (BWROG) Report, GE-NE-523-A71-0594, "Alternate BWR Feedwater Nozzle Inspection Requirements, August 1999," (TAC No. M94090), dated June 5, 1998

9.1.11 NRC Final SE related to "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75-A), EPRI Report TR-1012621, October 2005," dated March 16, 2006

9.1.12 NRC Final SE related to the "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75), EPRI Report TR-113932, October, 1999 (TAC NO. MA5012)," dated May 14, 2002

9.1.13 NRC Final SE related to the "BWR Reactor Vessel Shell Weld Inspection Recommendations (BWRVIP-05), EPRI Report TR-105697, September, 1995," dated July 28, 1998

9.1.14 NRC SE related to EPRI Topical Report TR-112657, Rev. B, Final Report, "Revised Risk-Informed Inservice Inspection Evaluation Procedure, July 1999," dated October 28, 1999

9.1.15 NRC Final SE related to the "License Renewal Safety Evaluation Report for the DNPS Units 2 and 3, and QCNPS Units 1 and 2," dated July 23, 2004

9.2 Industry References

9.2.1 ASME Boiler and Pressure Vessel Code, Section XI, Division 1, "Inservice Inspection of Nuclear Power Plant Components"

a. 2007 Edition through the 2008 Addenda (including Appendix VIII) (5th ISI Interval)
b. 2001 Edition through the 2003 Addenda (Subsection IWE)
c. 1998 Edition, No Addenda (1st CISI Interval Final)
d. 1995 Edition through the 1996 Addenda (4th ISI Interval)
e. 1992 Edition through the 1992 Addenda (1st CISI Interval Original)
f. 1989 Edition, No Addenda (3rd ISI Interval)

9.2.2 ASME Boiler and Pressure Vessel Code, Section V, "Nondestructive Examination," the 2007 Edition through the 2008 Addenda [The Edition and Addenda for ASME Section V are the same as the Edition and Addenda of ASME Section XI used for the inspection interval for both ISI and Non-ISI NDE examinations. Reference ASME Interpretation XI-1-89-02]

9.2.3 ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Rules For Construction of Nuclear Power Plant Components," the 2007 Edition through the 2008 Addenda

9.2.4 ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants," the 2004 Edition through the 2006 Addenda (Subsection ISTD)

9.2.5 NUREG-0619, dated November 1980, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking"

9.2.6 NUREG 0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping"

9.2.7 NUREG-1796, "Safety Evaluation Report Related to the License Renewal of the Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2"

9.2.8 NUREG-1801, "Generic Aging Lessons Learned"

9.2.9 Boiling Water Reactor Owners' Group (BWROG) Report GE-NE-523-A71-0594-A, Revision 1, "Alternate BWR Feedwater Nozzle Inspection Requirements," dated May 2000

9.2.10 Boiling Water Reactor Owners' Group (BWROG) Report GE-NE-523-A71-0594, "Alternate BWR Feedwater Nozzle Inspection Requirements," dated August 1999

9.2.11 BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75-A), EPRI Report TR-1012621, October 11, 2005
9.2.12 BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75), EPRI Report TR-113932, October 1999
9.2.13 Generic Letter 88-01, Revision 2, dated January 25, 1988, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping"
9.2.14 Generic Letter 88-01, Supplement 1, dated February 4, 1992, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping"
9.2.15 Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief From Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998
9.2.16 BWR Reactor Vessel Shell Weld Inspection Recommendations (BWRVIP-05), EPRI Report TR-105697, September, 1995
9.2.17 BWR Vessel and Internals Project Program Implementation Guide (BWRVIP-94), EPRI Report TR-1011702, dated December 2005
9.2.18 EPRI Topical Report TR-112657, Rev. B-A, Final Report, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," December 1999
9.2.19 EPRI Containment Inspection Program Guide (TR-110698-R1)
9.2.20 INPO Engineering Program Guide EPG-11, Inservice Inspection Program

9.3 <u>Licensee References</u>

9.3.1 DNPS Units 2 and 3 Updated Final Safety Analysis Report (UFSAR)
9.3.2 DNPS Units 2 and 3 Technical Specifications (TS)
9.3.3 DNPS Units 2 and 3 Technical Requirements Manual (TRM)
9.3.4 DNPS Units 2 and 3 ISI Classification Basis Document (DRE-483097-RP02), Fifth Ten-Year Inspection Interval
9.3.5 DNPS Units 2 and 3 ISI Selection Document (DRE-483097-RP04), Fifth Ten-Year Inspection Interval
9.3.6 Exelon Risk-Informed Inservice Inspection Evaluation (Final Report) for DNPS Units 2 and 3
9.3.7 DNPS Units 2 and 3, and QCNPS Units 1 and 2, "Operating License Renewal Application," January 3, 2003
9.3.8 DNPS Reactor Coolant Pressure Boundary Normal Makeup Calculation, XCE.040.0201
9.3.9 Calculation to Determine 80% of Primary Containment Remains Accessible for Examination, DRE03-0032 (EC343681) for Dresden Station Units 2 and 3
9.3.10 Procedures ER-AA-330, "Conduct of Inservice Inspection Activities," ER-AA-330-001, "Section XI Pressure Testing," ER-AA-330-002,

"Inservice Inspection of Welds and Components," ER-AA-330-003, "Visual Examination of Section XI Component Supports," ER-AA-330-004, "Visual Examination of Snubbers," ER-AA-330-007, "Visual Examination of Section XI Class MC Surfaces and Class CC Liners," ER-AA-330-009, "ASME Section XI Repair/Replacement Program," ER-AA-330-010, "Snubber Functional Testing," ER-AA-330-011, "Snubber Service Life Monitoring Program," ER-AA-335-004, "Manual Ultrasonic Measurement of Material Thickness," ER-AA-335-018, "Detailed, General, VT-1, VT-1C, VT-3, and VT-3C, Visual Examination of ASME Class MC and CC Containment Surfaces and Components," ER-AB-331, "BWR Internals Program Management," and ER-AB-331-101, "Evaluation for Thermal Aging/Neutron Embrittlement of Reactor Internals Components"

9.4 License Renewal References

9.4.1 AT 101522-01, ASME Section XI, Inservice Inspection License Renewal Commitments
9.4.2 AT 101522-07, BWR Stress Corrosion Cracking
9.4.3 AT 101522-05, BWR Feedwater Nozzle Commitments
9.4.4 AT 101522-49-04, Thermal Neutron Embrittlement Evaluation (CASS)
9.4.5 AT 101522-37, ASME Section XI, Subsection IWF