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Arkansas Nuclear One

2CAN011001

January 20, 2010

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

SUBJECT: Request for Alternative – Implementation of a Risk-Informed Inservice Inspection Program Based on ASME Code Case N-716
Arkansas Nuclear One, Unit 2
Docket No. 50-368
License No. NPF-6

Dear Sir or Madam:

Pursuant to 10 CFR 50.55a(a)(3)(i), Entergy Operations, Inc., (Entergy) hereby requests authorization to implement a risk-informed Inservice Inspection (RI-ISI) program at Arkansas Nuclear One, Unit 2 (ANO-2) based on the American Society of Mechanical Engineers (ASME) Code Case N-716, as documented in the attached Request for Alternative ANO2-ISI-005. ANO2-ISI-005 is being submitted in a template format. This template format is similar to the submittals the NRC Staff has approved for Waterford 3 and Grand Gulf.

In accordance with 10 CFR 50.55a(a)(3)(i), the proposed alternative to the referenced requirements may be approved by the NRC provided an acceptable level of quality and safety are maintained. Entergy believes the proposed alternative meets this requirement.

This request includes several new commitments that are summarized in Attachment 2.

Entergy requests approval of the proposed alternative by January 15, 2011, in order to support the spring 2011 refueling outage for ANO-2. Although this request is neither exigent nor emergency, your prompt review is requested.

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

Sincerely,

DBB/rwc

Attachment: 1. Request for Alternative ANO2-ISI-005
2. List of Regulatory Commitments

cc: Mr. Elmo E. Collins
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Attachment 1 to

2CAN011001

Request for Alternative ANO2-ISI-005

REQUEST FOR ALTERNATIVE
ENTERGY OPERATIONS, INC.
ARKANSAS NUCLEAR ONE – UNIT 2

REQUEST FOR ALTERNATIVE
ANO2-ISI-005

APPLICATION OF ASME CODE CASE N-716

RISK-INFORMED / SAFETY-BASED

INSERVICE INSPECTION PROGRAM PLAN

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ENTERGY OPERATIONS, INC.
ARKANSAS NUCLEAR ONE – UNIT 2

REQUEST FOR ALTERNATIVE
ANO2-ISI-005

1 INTRODUCTION

Arkansas Nuclear One – Unit 2 (ANO-2) is currently in the third inservice inspection (ISI) interval as defined by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Section XI Code for Inspection Program B. ANO-2 plans to implement a risk-informed / safety-based inservice inspection (RIS_B) program during the first inspection period of the fourth interval.

The ASME Section XI code of record for the fourth ISI interval at ANO-2 will be the 2001 Edition through 2003 Addenda for Examination Categories B-F, B-J, C-F-1, and C-F-2 Class 1 and 2 piping components.

The objective of this submittal is to request the use of the RIS_B process for the ISI of Class 1 and 2 piping. The RIS_B process used in this submittal is based upon ASME Code Case N-716, *Alternative Piping Classification and Examination Requirements, Section XI Division 1*, which is founded in large part on the RI-ISI process as described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A, *Revised Risk-Informed Inservice Inspection Evaluation Procedure*.

1.1 Relation to NRC Regulatory Guides 1.174 and 1.178

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guide (RG) 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis*, and RG 1.178, *An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping*. Additional information is provided in Section 3.4.2 relative to defense-in-depth.

1.2 Probabilistic Safety Assessment (PSA) Quality

The ANO-2 Level 1 PSA was initially developed in response to the NRC Generic Letter (GL) 88-20 on Individual Plant Examinations (IPEs). The ANO-2 IPE model was originally developed by ANO Design Engineering staff, Science Applications International Corporation, and ERIN Engineering and Research, Inc. The IPE was submitted to the NRC in 1992. The ANO-2 IPE consisted of the Level 1 PSA and back-end analysis (Level 2) consistent with the requirements of GL 88-20, *Individual Plant Examination for Severe Accident Vulnerabilities – 10 CFR 50.54(f)*. The NRC responded with a Safety Evaluation Report in a letter dated May 5, 1997, and approved the ANO-2 IPE results. The letter concluded that the ANO-2 IPE met the intent of GL 88-20; that is, the ANO-2 IPE process was capable of identifying the most likely severe accidents and severe accident vulnerabilities for ANO-2.

As part of the IPE development process, an expert panel review was performed on the results. This panel was composed of experienced personnel from ANO (Operations, Design Engineering, Training, and Licensing) supplemented by ERIN Engineering and Research, Inc. The comments from the IPE independent review were addressed, prior to its 1992 submittal to NRC.

Several PSA model updates have been completed on the ANO-2 PSA since the IPE was submitted. These were performed in order to maintain the PSA model reasonably consistent with the as-built, as-operated plant. The scope of the updates was based on review of results, plant input to the model, updated plant failure, and initiating event data as well as model enhancements. As part of each major update, in order to ensure adequacy of the updated model, an internal review of PSA model results is performed by utilizing an expert panel. The panel is typically composed of experienced personnel from various plant organizations, including Operations, System Engineering, Design Engineering, Safety Analysis, and PSA.

As described below, the ANO-2 PSA is more than adequate for this RI-ISI application. The PSA model used for this application has been updated to address peer review comments from a 2008 industry peer review. The findings have been evaluated. Most findings have been resolved and incorporated into the model used for this application or are documentation only concerns that have no impact on this application and are slated to be resolved in future updates. The remaining were reviewed relative to their potential impact on this application and are addressed in the information below. Entergy believes the ANO-2 PSA model fully supports the needs of this ISI submittal, as the internal flooding calculation core damage frequency (CDF) and large early release frequency (LERF) results for each scenario are well below the risk thresholds for the Code Case N-716.

History:

Industry Peer Review in 2002

An industry peer review of the ANO-2 PSA was conducted in 2002 on Revision 3p0 of the PSA model. The peer review concluded that there were several areas where the ANO-2 model needed improvement. The ANO-2 PSA model updates for Rev 3p01, 3p02, 4p00, and 4p01 (4p01 completed in October, 2006) addressed the significant Facts and Observations (F&O's) from this peer review.

Model Updates 3p01, 3p02, 4p00, and 4p01

The ANO-2 PSA Model has undergone several updates with substantial changes to most of the model components since the IPE submittal in 1992. New PSA software and utilities are now used for current revisions. A list of major changes from earlier ANO-2 PSA Model revisions to PSA Model 4p01 follows.

- Model Components Updated with Methodology Changes:
  - Human Reliability Analysis (HRA): Used the HRA Toolbox and developed HRA combinations
  - Common Cause Failure (CCF): Used the Entergy standard CCF (NUREG CR-5485) methodology and calculated the uncertainty parameters
  - Loss of Offsite Power: Updated with the most recent EPRI data and used the Entergy standard convolution method
  - Initiating Events: Updated with the most recent NRC / Industry data and developed fault-tree based initiating events

- New Model Components Added:
  - Anticipated Trip Without a Scram (ATWS): New ATWS event trees, developed the Reactor Protection System (RPS) / Diverse Scram System (DSS) / Emergency boration models and used NUREG/CR-5500 Vol. 11 for RPS reliability data
  - Intersystem Loss of Coolant Accident (ISLOCA): developed the ISLOCA package

Substantially Updated System Models:

- High Pressure Safety Injection (HPSI)
- Low Pressure Safety Injection / Shutdown Cooling (LPSI / SDC)
- Service Water
- Emergency Feedwater (EFW)
- AC/DC
- Instrument Air

- Other Significant Model Components Updated:
  - Accident Sequences / Top Logic
  - Plant-Specific Data
  - Reliability Database

The Rev 4p01 baseline total CDF was 3.020E-6/rx-yr, which was generated using PRAQuant with a quantification truncation limit of 1E-12/rx-yr. Recoveries are applied in this result and nominal test and maintenance unavailabilities are assumed.

Model Revision 4p02

In preparation for ANO-2's transition to the National Fire Protection Association (NFPA) 805 standard, a gap assessment of the ANO-2 PSA 4p01 model was completed. The gaps impacting the fire PSA were closed in the near term in order to meet the NFPA transition schedule. The ANO-2 Internal Events PSA model was updated (Rev 4p02 finished May, 2008) to meet the RG 1.200 Revision 1 standards. The baseline CDF for this model is 9.473E-07/rx-yr, which was generated using PRAQuant with a quantification truncation limit of 1E-12/rx-yr. Recoveries are applied in this result and nominal test and maintenance unavailabilities are assumed.

RG 1.200 Peer Review, July 2008

A peer review on the Rev 4p02 ANO-2 PSA Model was performed in July, 2008. Some of the significant F&Os have been addressed and resolved. The remaining open gaps are dispositioned for the RI-ISI application as detailed in Appendix 1.

Internal Flooding Model

The ANO-2 Internal Flooding Analysis (IFA) was significantly upgraded to meet the requirements of RG 1.200, Revision 1 in 2009. This analysis was used in the subject RI-ISI evaluation to determine the High Safety-Significant (HSS) scope and as an input to Low Safety-Significant (LSS) scope Conditional Core Damage Probability (CCDP) values used in the risk impact assessment. This analysis is a substantial improvement over the previous IPE version. As an example, the IPE IFA conservatively used a 1E-6 screening value, and no scenarios resulted in CDF higher than the screening value. The current IFA has hundreds of quantified scenarios, with CDF ranging from the 1E-7 range to the 1E-12 range. Many of the scenarios have a CDF lower than the quantification truncation value used (1E-12).

Large Early Release Frequency (LERF) Model

The ANO-2 LERF model was prepared based on the Westinghouse LERF model (WCAP-16341-P) with specific enhancements and changes to reflect the plant specific features. This LERF model was linked with the Rev 4p02 Internal Events CDF model to generate the LERF. The baseline LERF for this model is 1.08E-07/rx-yr; the quantification truncation used was 1.00E-12.

2 PROPOSED ALTERNATIVE TO CURRENT ISI PROGRAMS

2.1 ASME Section XI

ASME Section XI Examination Categories B-F, B-J, C-F-1, and C-F-2 currently contain requirements for the nondestructive examination (NDE) of Class 1 and 2 piping components, except as amended by the following alternative requirements:

- A risk-informed inservice inspection (RI-ISI) program based on ASME Code Case N-578 was approved for use at ANO-2 by the NRC on December 29, 1998.

The alternative RIS_B Program for piping is described in Code Case N-716. The RIS_B Program will be substituted for the current program for Class 1 and 2 piping (Examination Categories B-F, B-J, C-F-1 and C-F-2) in accordance with 10 CFR 50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety. Other non-related portions of the ASME Section XI Code will be unaffected.

2.2 Augmented Programs

The impact of the RIS_B application on the various plant augmented inspection programs listed below were considered. This section documents only those plant augmented inspection programs that address common piping with the RIS_B application scope (e.g., Class 1 and 2 piping).

- The plant augmented inspection program for high energy line breaks (HELB) and moderate energy line breaks (MELB) outside containment, implemented in accordance with ANO Upper Level Document (ULD) ULD-0-TOP-07, *HELB/MELB Topical ULD*, is not affected or changed by the RIS_B Program.

- The plant augmented inspection programs previously implemented in response to NRC Bulletins 88-08, *Thermal Stresses in Piping Connected to Reactor Coolant Systems*, and 88-11, *Pressurizer Surge Line Thermal Stratification*, were subsumed by the RI-ISI Program since the thermal fatigue concerns addressed by these bulletins were explicitly considered in the application of the RI-ISI process. Since the RI-ISI and RIS_B degradation mechanism criterion is identical, the plant augmented inspection programs are subsumed by the new RIS_B Program.

- The plant augmented inspection program for flow accelerated corrosion (FAC) per Generic Letter (GL) 89-08, *Erosion/Corrosion-Induced Pipe Wall Thinning*, is relied upon to manage this damage mechanism but is not otherwise affected or changed by the RIS_B Program.

- A plant augmented inspection program is being implemented at ANO-2 in response to IE Bulletin 79-17, *Pipe Cracks in Stagnant Borated Water Systems at PWR Plants*. The intergranular stress corrosion cracking concern addressed by this bulletin was explicitly considered in the application of the RIS_B process and is subsumed by the RIS_B Program.

- A plant augmented inspection program is being implemented at ANO-2 in response to MRP-139, *Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines*. The requirements of MRP-139 will be used for the inspection and management of primary water stress corrosion cracking (PWSCC) susceptible welds and will supplement the RIS_B Program selection process. The RIS_B Program will not be used to eliminate any MRP-139 requirements.

- ANO-2 is in the process of evaluating MRP-146, *Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines*. Results will be incorporated into the RIS_B Program, as warranted.

3 RISK-INFORMED / SAFETY-BASED ISI PROCESS

The process used to develop the RIS_B Program conformed to the methodology described in Code Case N-716 and consisted of the following steps:

- Safety Significance Determination
- Failure Potential Assessment
- Element and NDE Selection
- Risk Impact Assessment
- Implementation Program
- Feedback Loop

3.1 Safety Significance Determination

The systems assessed in the RIS_B Program are provided in Table 3.1. The piping and instrumentation diagrams and additional plant information, including the existing plant ISI Program, were used to define the piping system boundaries.

Per Code Case N-716 requirements, piping welds are assigned safety-significance categories, which are used to determine the treatment requirements. HSS welds are determined in accordance with the requirements below. LSS welds include all other Class 2, 3, or Non-Class welds.

(1) Class 1 portions of the reactor coolant pressure boundary (RCPB), except as provided in 10 CFR 50.55a(c)(2)(i) and (c)(2)(ii);

(2) Applicable portions of the SDC pressure boundary function. That is, Class 1 and 2 welds of systems or portions of systems needed to utilize the normal SDC flow path either:

   (a) As part of the RCPB from the reactor pressure vessel (RPV) to the second isolation valve (i.e., farthest from the RPV) capable of remote closure or to the containment penetration, whichever encompasses the larger number of welds; or

   (b) Other systems or portions of systems from the RPV to the second isolation valve (i.e., farthest from the RPV) capable of remote closure or to the containment penetration, whichever encompasses the larger number of welds;

(3) That portion of the Class 2 feedwater system [> 4 inch nominal pipe size (NPS)] of pressurized water reactors (PWRs) from the steam generator to the outer containment isolation valve;

(4) Piping within the break exclusion region (> NPS 4) for high-energy piping systems as defined by the Owner. This may include Class 3 or Non-Class piping; and

(5) Any piping segment whose contribution to CDF is greater than 1E-06 (and per NRC feedback on the Grand Gulf and DC Cook RIS_B pilot applications, 1E-07 for LERF) based upon a plant-specific PSA of pressure boundary failures (e.g., pipe whip, jet impingement, spray, inventory losses). This may include Class 3 or Non-Class piping.

3.2 Failure Potential Assessment

Failure potential estimates were generated utilizing industry failure history, plant-specific failure history, and other relevant information. These failure estimates were determined using the guidance provided in EPRI TR-112657 (i.e., the EPRI RI-ISI methodology), with the exception of the deviation discussed below.

Table 3.2 summarizes the failure potential assessment by system for each degradation mechanism that was identified as potentially operative.

A deviation to the EPRI RI-ISI methodology has been implemented in the failure potential assessment for ANO-2. Table 3-16 of EPRI TR-112657 contains criteria for assessing the potential for thermal stratification, cycling, and striping (TASCS). Key attributes for horizontal or slightly sloped piping greater than NPS 1 include:

1. The potential exists for low flow in a pipe section connected to a component allowing mixing of hot and cold fluids; or

2. The potential exists for leakage flow past a valve, including in-leakage, out-leakage and cross-leakage allowing mixing of hot and cold fluids; or

3. The potential exists for convective heating in dead-ended pipe sections connected to a source of hot fluid; or

4. The potential exists for two phase (steam/water) flow; or

5. The potential exists for turbulent penetration into a relatively colder branch pipe connected to header piping containing hot fluid with turbulent flow

   AND

   $\Delta T > 50°F$,

   AND

   Richardson Number > 4 (this value predicts the potential buoyancy of a stratified flow)

These criteria, based on meeting a high cycle fatigue endurance limit with the actual ΔT assumed equal to the greatest potential differential temperature (ΔT) for the transient, will identify locations where stratification is likely to occur, but allows for no assessment of severity. As such, many locations will be identified as subject to TASCS where no significant potential for thermal fatigue exists. The critical attribute missing from the existing methodology that would allow consideration of fatigue severity is a criterion that addresses the potential for fluid cycling. The impact of this additional consideration on the existing TASCS susceptibility criteria is presented below.

- **Turbulent Penetration TASCS**

Turbulent penetration typically occurs in lines connected to piping containing hot flowing fluid. In the case of downward sloping lines that then turn horizontal, significant top-to-bottom cyclic ΔTs can develop in the horizontal sections if the horizontal section is less than about 25 pipe diameters from the reactor coolant piping. Therefore, TASCS is considered for this configuration.

For upward sloping branch lines connected to the hot fluid source that turn horizontal or in horizontal branch lines, natural convective effects combined with effects of turbulence penetration will keep the line filled with hot water. If there is no potential for in-leakage towards the hot fluid source from the outboard end of the line, this will result in a well-mixed fluid condition where significant top-to-bottom ΔTs will not occur. Therefore, TASCS is not considered for these configurations. Even in fairly long lines, where some heat loss from the outside of the piping will tend to occur and some fluid stratification may be present, there is no significant potential for cycling as has been observed for the in-leakage case. The effect of TASCS will not be significant under these conditions and can be neglected.

- **Low flow TASCS**

In some situations, the transient startup of a system (e.g., shutdown cooling suction piping) creates the potential for fluid stratification as flow is established. In cases where no cold fluid source exists, the hot flowing fluid will fairly rapidly displace the cold fluid in stagnant lines, while fluid mixing will occur in the piping further removed from the hot source and stratified conditions will exist only briefly as the line fills with hot fluid. As such, since the situation is transient in nature, it can be assumed that the criteria for thermal transients (TT) will govern.

- **Valve leakage TASCS**

Sometimes a very small leakage flow of hot water can occur outward past a valve into a line that is relatively colder, creating a significant temperature difference. However, since this is generally a "steady-state" phenomenon with no potential for cyclic temperature changes, the effect of TASCS is not significant and can be neglected.

- **Convection Heating TASCS**

Similarly, there sometimes exists the potential for heat transfer across a valve to an isolated section beyond the valve, resulting in fluid stratification due to natural convection. However, since there is no potential for cyclic temperature changes in this case, the effect of TASCS is not significant and can be neglected.

In summary, these additional considerations for determining the potential for thermal fatigue as a result of the effects of TASCS provide an allowance for considering cycle severity. The above criteria have previously been submitted by EPRI to the NRC for generic approval [letters dated February 28, 2001, and March 28, 2001, from P.J. O'Regan (EPRI) to Dr. B. Sheron (USNRC), *Extension of Risk-Informed Inservice Inspection Methodology*]. The methodology used in the ANO-2 RIS_B application for assessing TASCS potential conforms to these updated criteria. Final Materials Reliability Program (MRP) guidance on the subject of TASCS will be incorporated into the ANO-2 RIS_B application, as warranted. It should be noted that the NRC has granted approval for RI-ISI relief requests incorporating these TASCS criteria at several facilities, including Comanche Peak (NRC letter dated September 28, 2001) and South Texas Project (NRC letter dated March 5, 2002).

3.3 Element and NDE Selection

Code Case N-716 and lessons learned from the Grand Gulf and DC Cook RIS_B pilot applications provide criteria for identifying the number and location of required examinations. Ten percent of the HSS welds shall be selected for examination as follows:

(1) Examinations shall be prorated equally among systems to the extent practical, and each system shall individually meet the following requirements:

   (a) A minimum of 25% of the population identified as susceptible to each degradation mechanism and degradation mechanism combination shall be selected.

   (b) If the examinations selected above exceed 10% of the total number of HSS welds, the examinations may be reduced by prorating among each degradation mechanism and degradation mechanism combination, to the extent practical, such that at least 10% of the HSS population is inspected.

   (c) If the examinations selected above are not at least 10% of the HSS weld population, additional welds shall be selected so that the total number selected for examination is at least 10%.

(2) At least 10% of the Reactor Coolant Pressure Boundary (RCPB) welds shall be selected.

(3) For the RCPB, at least two-thirds of the examinations shall be located between the inside first isolation valve (IFIV) (i.e., isolation valve closest to the RPV) and the RPV.

(4) A minimum of 10% of the welds in that portion of the RCPB that lies outside containment (OC) shall be selected.

(5) A minimum of 10% of the welds within the break exclusion region (BER) shall be selected.

In contrast to a number of RI-ISI Program applications where the percentage of Class 1 piping locations selected for examination has fallen substantially below 10%, Code Case N-716 mandates that 10% be chosen. A brief summary is provided below, and the results of the selections are presented in Table 3.3. Section 4 of EPRI TR-112657 was used as guidance in determining the examination requirements for these locations.

Unit	Class 1 Welds(1)		Class 2 Welds(2)		NNS Welds(3)		All Piping Welds(4)	
	Total	Selected	Total	Selected	Total	Selected	Total	Selected
2	692	74	1,754	7	0	0	2,446	81

Notes

(1) Includes all Category B-F and B-J locations. All Class 1 piping weld locations are HSS.

(2) Includes all Category C-F-1 and C-F-2 locations. Of the 1,754 Class 2 piping weld locations, 97 are HSS and the remaining 1,657 are LSS.

(3) There is no Class 3 or non-nuclear safety (NNS) piping weld locations determined to be HSS.

(4) Regardless of safety significance, Class 1 and 2 in-scope piping components will continue to be pressure tested as required by the ASME Section XI Program. VT-2 visual examinations are scheduled in accordance with the station's pressure test program that remains unaffected by the RIS_B Program.

3.3.1 Additional Examinations

If the flaw is original construction or otherwise is acceptable, Code rules do not require any additional inspections. Any unacceptable flaw will be evaluated per the requirements of ASME Code Section XI, IWB-3500 and/or IWB-3600. As part of performing evaluation to IWB-3600, the degradation mechanism that is responsible for the flaw will be determined and accounted for in the evaluation. The process for ordinary flaws is to perform the evaluation using ASME Section XI. If the flaw meets the criteria, then it is noted and appropriate successive examinations scheduled. If the nature and type of the flaw is service-induced, then similar systems or trains will be examined. If the flaw is found unacceptable for continued operation, it will be repaired in accordance with IWA-4000 and/or applicable ASME Section XI Code Cases. The need for extensive root cause analysis beyond that required for IWB-3600 evaluation will be dependent on practical considerations (i.e., the practicality of performing additional non-destructive examination (NDE) or removing the flaw for further evaluation during the outage). The NRC is involved in the process at several points. For preemptive weld overlays, a relief request in accordance with 10 CFR 50.55a(a)(3) is usually required for design and installation. Should a flaw be discovered during an examination, a notification in accordance with 10 CFR 50.72 or 10 CFR 50.73 may be required. IWB-3600 requires the evaluation to be submitted to the NRC. Finally, the evaluation will be documented in the corrective action program and the Owner submittals required by Section XI.

The evaluation will include whether other elements in the segment or additional segments are subject to the same root cause conditions. Additional examinations will be performed on those elements with the same root cause conditions or degradation mechanisms. The additional examinations will include HSS elements up to a number equivalent to the number of elements required to be inspected during the current outage. If unacceptable flaws or relevant conditions are again found similar to the initial problem, the remaining elements identified as susceptible will be examined during the current outage. No additional examinations need be performed if there are no additional elements identified as being susceptible to the same root cause conditions.

3.3.2 Program Relief Requests

An attempt has been made to select RIS_B locations for examination such that a minimum of > 90% coverage (i.e., Code Case N-460 criteria) is attainable. However, some limitations will not be known until the examination is performed since some locations may be examined for the first time by the specified techniques.

In instances where locations at the time of the examination fail to meet the > 90% coverage requirement, the process outlined in 10 CFR 50.55a will be followed.

Per Footnote 3 of Table 1 of Code Case N-716, when the required examination volume or area cannot be examined due to interference by another component or part geometry, limited examinations shall be evaluated for acceptability. Acceptance of limited examinations or volumes shall not invalidate the results of the change-in-risk evaluation (Paragraph 5 of Code Case N-716). The change in risk evaluation of Code Case N-716 is consistent with previous RI-ISI applications and meets RG 1.174 change-in-risk acceptance criteria. Areas with acceptable limited examinations, and their bases, shall be documented.

Consistent with previously approved RI-ISI submittals, ANO-2 will calculate coverage and use additional examinations or techniques in the same manner used for traditional Section XI examinations. Experience has shown this process to be weld-specific (e.g., joint configuration). As such, the effect on risk, if any, will not be known until that time. Relief requests will be submitted per the guidance of 10 CFR 50.55a(g)(5)(iv) within one (1) year after the end of the interval.

The request for alternative pertaining to the use of Code Case N-578, will be withdrawn for use at ANO-2 upon NRC approval of the RIS_B Program submittal.

3.4 Risk Impact Assessment

The RIS_B Program development has been conducted in accordance with RG 1.174 and the requirements of Code Case N-716. The risk from implementation of this program is expected to remain neutral or decrease when compared to that estimated from current requirements.

This evaluation categorized segments as HSS or LSS in accordance with Code Case N-716, and then determined what inspection changes are proposed for each system. The changes include changing the number and location of inspections and in many cases improving the effectiveness of the inspection to account for the findings of the RIS_B degradation mechanism assessment. For example, examinations of locations subject to thermal fatigue will be conducted on an expanded volume and will be focused to enhance the probability of detection (POD) during the inspection process.

3.4.1 Quantitative Analysis

Code Case N-716 has adopted the EPRI TR-112657 process for risk impact analyses whereby limits are imposed to ensure that the change in risk of implementing the RIS_B Program meets the requirements of RG 1.174 and 1.178. The EPRI criterion requires that the cumulative change in CDF and LERF be less than 1E-07 and 1E-08 per year per system, respectively.

For LSS welds, CCDP and conditional large early release probability (CLERP) values of 1E-4 and 1E-5 are generally conservatively used, unless pipe segments in the plant internal flooding study are found with higher values. For the ANO-2 RIS_B application, CCDP and CLERP values greater than 1E-4 and 1E-5 have been used for LSS welds to bound plant internal flooding study results. These values used for CCDP and CLERP were determined based on results from the plant internal flooding study and have been conservatively applied as an upper bound for all LSS welds in the applicable system (see table below).

With respect to assigning failure potential for LSS piping, the criteria are defined by Table 3 of the Code Case. That is, those locations identified as susceptible to FAC (or another mechanism and also susceptible to water hammer) are assigned a high failure potential. Those locations susceptible to thermal fatigue, erosion-cavitation, corrosion or stress corrosion cracking are assigned to a medium failure potential and those locations that are identified as not susceptible to degradation are assigned a low failure potential.

In order to streamline the risk impact assessment, a review was conducted to verify that the LSS piping was not susceptible to FAC or water hammer. This review was conducted similar to that performed for a traditional RI-ISI application. Thus, the high failure potential category is not applicable to LSS piping. In lieu of conducting a formal degradation mechanism evaluation for all LSS piping (e.g., to determine if thermal fatigue is applicable), these locations were conservatively assigned to the medium failure potential category ("Assume Medium" in Table 3.4-1) for use in the change-in-risk assessment. Experience with previous industry RI-ISI applications shows this to be conservative.

ANO-2 has conducted a risk impact analysis per the requirements of Section 5 of Code Case N-716 that is consistent with the "Simplified Risk Quantification Method" described in Section 3.7 of EPRI TR-112657. The analysis estimates the net change in risk due to the positive and negative influences of adding and removing locations from the inspection program.

The CCDP and CLERP values used to assess risk impact were estimated based on pipe break location. Based on these estimated values, a corresponding consequence rank was assigned per the requirements of EPRI TR-112657 and upper bound threshold values were used as provided below. Consistent with the EPRI risk-informed methodology, the upper bound for all HSS RCPB break locations that fall within the high consequence rank range was based on the highest CCDP value obtained (i.e., large Loss of Coolant Accident (LOCA) for breaks inside containment). For breaks outside containment, only the main feedwater (MFW) system is in the HSS scope and its CCDP falls into the high consequence rank range as shown in the table below. In addition, as described above, certain LSS pipe segments were found with CCDP in the high consequence range. In each case, the highest CCDP found for a system was conservatively applied to all LSS piping in the system.

CCDP and CLERP Values Based on Break Location					
Break Location Designation	**Estimated**		**Consequence Rank**	**Upper Bound**	
	CCDP	**CLERP**		**CCDP**	**CLERP**
LOCA	4.0E-04	4.0E-05	**HIGH**	4.0E-04	4.0E-05
RCPB pipe breaks that result in a loss of coolant accident - The highest CCDP for Large LOCA (4E-4) was used (0.1 margin used for CLERP). Unisolated RCPB piping of all sizes.					
ILOCA(1)	1.2E-06	1.2E-07	**MEDIUM**	1.0E-04	1.0E-05
Isolable LOCA (1 open valve) - RCPB pipe breaks that result in an isolable LOCA - Calculated based on Large LOCA CCDP of 4E-4 and valve fail to close probability of ~3E-3 (0.1 margin used for CLERP). Between 1st and 2nd isolation valve on charging, letdown and pressurizer relief.					
PLOCA(1)	<1E-06	<1E-07	**MEDIUM**	1.0E-04	1.0E-05
Potential LOCA (1 closed valve) - RCPB pipe breaks that result in a potential LOCA - Calculated based on Large LOCA CCDP of 4E-4 and valve rupture probability of ~1E-3 (0.1 margin used for CLERP). Piping between 1st and 2nd isolation valves on safety injection, alternate charging, drain lines on RC					
PPLOCA(1)	<1E-06	<1E-07	**MEDIUM**	1.0E-04	1.0E-05
Potential LOCA (2 valves) - Class 2 potential LOCA in shutdown cooling piping downstream of 2nd isolation valve (suction motor operated valves (MOVs)) and upstream of 2nd isolation valve (return lines check valves). LOCA CCDP and failure of 2 MOVs to close on demand (3E-4) is judged to be appropriate for lines inside containment (0.1 margin used for CLERP)					
SLB	<1E-06	<1E-07	**MEDIUM**	1.0E-04	1.0E-05
Main steam and feedwater breaks %T5-A, %T5-B, %T5-C (0.1 margin used for CLERP). Used for MFW HSS scope inside containment and supports Class 2 LSS for MS scope below.					
MFW-OC(2)	2.0E-03	2.0E-04	**HIGH**	2.0E-03	2.0E-04
MFW HSS scope outside containment based on internal flooding analysis. (0.1 margin sued for CLERP).					
Class 2 LSS	1.0E-04	1.0E-05	**MEDIUM**	1.0E-04	1.0E-05
Class 2 pipe breaks that occur in the remaining system piping designated as low safety significant - Estimated based on upper bound for Medium Consequence. Used for CH, MS and SW LSS.					
CS LSS(3)	3.0E-03	3.0E-04	**HIGH**	3.0E-03	3.0E-04
Class 2 CS LSS scope pipe breaks based on internal flood analysis (0.1 margin used for CLERP)					
EF LSS(3)	6.0E-02	6.0E-03	**HIGH**	6.0E-02	6.0E-03
Class 2 EF LSS scope pipe breaks based on internal flood analysis (0.1 margin used for CLERP)					
SI LSS(3)	9.0E-04	9.0E-05	**HIGH**	9.0E-04	9.0E-05
Class 2 SI LSS scope pipe breaks based on internal flood analysis (0.1 margin used for CLERP)					

(System acronyms are described in Table 3.1)

Notes

(1) These are RCPB pipe segments beyond 1st isolation valve. PLOCA = potential LOCA (1 closed valve), PPLOCA = potential LOCA (2 closed valves), ILOCA = isolable LOCA (1 open valve). These are not modeled in PSA, but are estimated based on valve failure probabilities. Although the calculated CCDP and CLERP values for PLOCA break locations fall in the "Low" consequence rank range, a "Medium" consequence rank is conservatively used for risk impact.

(2) CCDP is from Internal Flooding Analysis for breaks outside containment

(3) CCDP from Internal Flooding Analysis. Found for certain segments and assigned to all the system scope. These are LSS from a risk perspective and the CCDP is likely conservative for this reason.

The likelihood of pressure boundary failure (PBF) is determined by the presence of different degradation mechanisms and the rank is based on the relative failure probability. The basic likelihood of PBF for a piping location with no degradation mechanism present is given as x_o and is expected to have a value less than 1E-08. Piping locations identified as medium failure potential have a likelihood of $20x_o$. These PBF likelihoods are consistent with References 9 and 14 of EPRI TR-112657. In addition, the analysis was performed both with and without taking credit for enhanced inspection effectiveness due to an increased POD from application of the RIS_B approach.

Table 3.4-1 presents a summary of the RIS_B Program versus 1986 ASME Section XI Code Edition program requirements on a "per system" basis. The presence of FAC and PWSCC was adjusted for in the quantitative analysis by excluding their impact on the failure potential rank. The exclusion of the impact of FAC and PWSCC on the failure potential rank and therefore in the determination of the change in risk is appropriate, because FAC and PWSCC are damage mechanisms managed by separate, independent plant augmented inspection programs. The RIS_B Program credits and relies upon these plant augmented inspection programs to manage these damage mechanisms. The plant FAC and PWSCC Programs will continue to determine where and when examinations shall be performed. Hence, since the number of FAC and PWSCC examination locations remains the same "before" and "after" and no delta exist, there is no need to include the impact of FAC and PWSCC in the performance of the risk impact analysis.

As indicated in the following table, this evaluation has demonstrated that unacceptable risk impacts will not occur from implementation of the RIS_B Program, and satisfies the acceptance criteria of RG 1.174 and Code Case N-716. In addition, sensitivity cases with the estimated CCDP and CLERP values in the above table show that the change is risk is similar and meets the acceptance criteria. Also, for cases where the RIS_B selections exceeded SXI selections in Table 3.4-1, they were set equal to SXI to confirm that the use of conservative CCDP and CLERP are not non-conservative relative to meeting the acceptance criteria.

ANO-2 Risk Impact Results				
System	**With POD Credit**		**Without POD Credit**	
	Delta CDF	**Delta LERF**	**Delta CDF**	**Delta LERF**
CH - Chemical & Volume Control	-6.76E-10	-6.76E-11	-3.24E-10	-3.24E-11
MFW - Main Feedwater	-4.85E-11	-4.85E-12	1.15E-11	1.15E-12
RCS - Reactor Coolant	-7.70E-10	-7.70E-11	-2.34E-10	-2.34E-11
SI - Safety Injection	5.77E-09	5.77E-10	6.06E-09	6.06E-10
CS - Containment Spray	3.60E-09	3.60E-10	3.60E-09	3.60E-10
EF - Emergency Feedwater	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MS - Main Steam	1.30E-10	1.30E-11	1.30E-10	1.30E-11
SW - Service Water	4.00E-11	4.00E-12	4.00E-11	4.00E-12
Total	**8.05E-09**	**8.05E-10**	**9.29E-09**	**9.29E-10**

3.4.2 Defense-in-Depth

The intent of the inspections mandated by ASME Section XI for piping welds is to identify conditions such as flaws or indications that may be precursors to leaks or ruptures in a system's pressure boundary. Currently, the process for selecting inspection locations is based upon structural discontinuity and stress analysis results. As depicted in ASME White Paper 92-01-01 Rev. 1, *Evaluation of Inservice Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds*, this method has been ineffective in identifying leaks or failures. EPRI TR-112657 and Code Case N-716 provide a more robust selection process founded on actual service experience with nuclear plant piping failure data.

This process has two key independent ingredients; that is, a determination of each location's susceptibility to degradation and secondly, an independent assessment of the consequence of the piping failure. These two *ingredients* assure defense-in-depth is maintained. First, by evaluating a location's susceptibility to degradation, the likelihood of finding flaws or indications that may be precursors to leaks or ruptures is increased. Secondly, a generic assessment of high-consequence sites has been determined by Code Case N-716 supplemented by plant-specific evaluations, thereby requiring a minimum threshold of inspection for important piping whose failure would result in a LOCA or BER break. Finally, Code Case N-716 requires that any piping on a plant-specific basis that has a contribution to CDF of greater than 1E-06 (or 1E-07 for LERF) be included in the scope of the application. ANO-2 did not identify any such piping.

All locations within the Class 1, 2, and 3 pressure boundaries will continue to be pressure tested in accordance with the Code, *regardless* of its safety significance.

4 IMPLEMENTATION AND MONITORING PROGRAM

Upon approval of the RIS_B Program, procedures that comply with the guidelines described in EPRI TR-112657 will be prepared to implement and monitor the program. The new program will be implemented in the fourth ISI interval. No changes to the Technical Specifications or Updated Final Safety Analysis Report are necessary for program implementation.

The applicable aspects of the ASME Code not affected by this change will be retained, such as inspection methods, acceptance guidelines, pressure testing, corrective measures, documentation requirements, and quality control requirements. Existing ASME Section XI program implementing procedures will be retained and modified to address the RIS_B process, as appropriate.

The monitoring and corrective action program will contain the following elements**:**

A. Identify

B. Characterize

C. (1) Evaluate, determine the cause and extent of the condition identified

   (2) Evaluate, develop a corrective action plan or plans

D. Decide

E. Implement

F. Monitor

G. Trend

The RIS_B Program is a living program requiring feedback of new relevant information to ensure the appropriate identification of HSS piping locations. As a minimum, this review will be conducted on an ASME period basis. In addition, significant changes may require more frequent adjustment as directed by NRC Bulletin or Generic Letter requirements, or by industry and plant-specific feedback.

For preservice examinations, ANO-2 will follow the rules contained in Section 3.0 of Code Case N-716. Welds classified HSS require preservice inspection. The examination volumes, techniques, and procedures shall be in accordance with Table 1 of Code Case N-716. Welds classified as LSS do not require preservice inspection.

5 PROPOSED ISI PROGRAM PLAN CHANGE

A comparison between the RIS_B Program and ASME Section XI 2001 Code Edition program requirements for in-scope piping is provided in Table 5.1.

ANO-2 intends to start implementing the RIS_B Program during the plant's first period of the fourth inspection interval. The fourth ISI interval will implement 100% of the inspection locations selected for examination per the RIS_B Program. Examinations will be performed such that the period percentage requirements of ASME Section XI are met.

6 REFERENCES/DOCUMENTATION

1. USNRC Safety Evaluation pertaining to the use of ASME Code Case N-578, dated December 29, 1998 (Letter 2CNA129805)

2. EPRI TR-112657, Revised *Risk-Informed Inservice Inspection Evaluation Procedure*, Rev. B-A

3. ASME Code Case N-716, *Alternative Piping Classification and Examination Requirements, Section XI Division 1*

4. Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis*

5. Regulatory Guide 1.178, *An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping*

Supporting Onsite Documentation

1. Structural Integrity Calculation 0800567.301, *Degradation Mechanism Evaluation for ANO-2*, Revision 0

2. Structural Integrity Calculation 0800567.302, *N-716 Evaluation of Arkansas Nuclear One – Unit 2*, Revision 0

Table 3.1
N-716 Safety Significance Determination

System Description	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: MFW	BER	CDF > 1E-6	High	Low
CH - Chemical & Volume Control	113	x					x	
	5							x
MFW - Main Feedwater	64			x			x	
RCS - Reactor Coolant	316	X					x	
SI - Safety Injection	240	x	x				x	
	23	x					x	
	33		x				x	
	1272							x
CS - Containment Spray	173							x
EF - Emergency Feedwater	94							x
MS - Main Steam	89							x
SW - Service Water	24							x
SUMMARY RESULTS FOR ALL SYSTEMS	240	x	x				x	
	452	x					x	
	33		x				x	
	64			x			x	
	1657							x
TOTALS	**2446**							

Table 3.2

Failure Potential Assessment Summary

System[1]	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive	
	TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC
CH		x									
MFW	x										
RCS	x	x				x					
SI	x		x			x					
CS											
EF											
MS											
SW											

Notes

1. Systems are described in Table 3.1.
2. A degradation mechanism assessment was not performed on low safety significant piping segments. This includes the CS, EF, MS, and SW systems in their entirety, as well as portions of the CH and SI systems.
3. IGSCC = Intergranular Stress Corrosion Cracking; TGSCC = Transgranular Stress Corrosion Cracking; ECSCC = External Chloride Stress Corrosion Cracking; MIC = Microbiologically-Influenced Corrosion; PIT = Pitting; CC = Crevice Corrosion; E-C = Erosion-Cavitation.

Tale 3.3 N-716 Element Selections								
System[1]	**Weld Count**		**N716 Selction Considerations**					**Selections**
	HSS	**LSS**	**DMs[2]**	**RCPB**	**RCPB (IFIV)**	**RCPB (OC)**	**BER**	
CH	32		TT	**x**	**x**			10
CH	3		TT	**x**				0
CH	61		None	**x**	**x**			2
CH	17		None	**x**				0
CH		5						0
MFW	6		TASCS					6
MFW	58		None					1
RCS	1		PWSCC	**x**	**x**			0
RCS	1		PWSCC,TT	**x**	**x**			0
RCS	9		TASCS	**x**	**x**			3
RCS	17		TASCS,TT	**x**	**x**			8
RCS	9		TT	**x**	**x**			2
RCS	6		TT	**x**				2
RCS	225		None	**x**	**x**			17
RCS	48		None	**x**				0
SI	5		IGSCC	**x**				2
SI	1		PWSCC	**x**	**x**			0
SI	9		TASCS	**x**	**x**			8
SI	44		None	**x**	**x**			17
SI	204		None	**x**				3
SI	33		None					0
SI		1272						0
CS		173						0
EF		94						0
MS		89						0
SW		24						0
Summary Results All Systems	41		TT	**x**	**x**			12
	9		TT	**x**				2
	6		TASCS					6
	2		PWSCC	**x**	**x**			0
	1		PWSCC,TT	**x**	**x**			0
	18		TASCS	**x**	**x**			11
	17		TASCS,TT	**x**				8
	5		IGSCC	**x**				2
	332		None	**x**	**x**			36
	267		None	**x**				3
	91		None					1
		1657						0
Totals	**789**	**1657**						**81**

Notes

1. Systems are described in Table 3.1.
2. Damage Mechanisms

Table 3.4-1
Risk Impact Analysis Results

System[1]	Safety Significance	Break Location	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank	SXI	RIS_B	Delta	w/POD	w/o POD	w/POD	w/o POD
CH	High	LOCA	TT	Medium	2	10	8	-6.72E-10	-3.20E-10	-6.72E-11	-3.20E-11
CH	High	ILOCA	TT	Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CH	High	LOCA	None	Low	0	2	2	-4.00E-12	-4.00E-12	-4.00E-13	-4.00E-13
CH	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CH	High	ILOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CH	Low	Class 2 LSS		Assume Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CH Total								**-6.76E-10**	**-3.24E-10**	**-6.76E-11**	**-3.24E-11**
MFW	High	SLB	TASCS	Medium	3	6	3	-9.00E-11	-3.00E-11	-9.00E-12	-3.00E-12
MFW	High	SLB	None	Low	4	1	-3	1.50E-12	1.50E-12	1.50E-13	1.50E-13
MFW	High	MFW-OC	None	Low	4	0	-4	4.00E-11	4.00E-11	4.00E-12	4.00E-12
MFW Total								**-4.85E-11**	**1.15E-11**	**-4.85E-12**	**1.15E-12**
RCS	High	LOCA	(PWSCC)	Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RCS	High	LOCA	(PWSCC), TT	Medium	1	0	-1	2.40E-11	4.00E-11	2.40E-12	4.00E-12
RCS	High	LOCA	TASCS	Medium	0	3	3	-2.16E-10	-1.20E-10	-2.16E-11	-1.20E-11
RCS	High	LOCA	TASCS,TT	Medium	4	8	4	-4.80E-10	-1.60E-10	-4.80E-11	-1.60E-11
RCS	High	LOCA	TT	Medium	1	2	1	-1.20E-10	-4.00E-11	-1.20E-11	-4.00E-12
RCS	High	PLOCA	TT	Medium	2	2	0	-2.40E-11	0.00E+00	-2.40E-12	0.00E+00
RCS	High	LOCA	None	Low	38	15	-23	4.60E-11	4.60E-11	4.60E-12	4.60E-12
RCS	High	ILOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RCS	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RC Total								**-7.70E-10**	**-2.34E-10**	**-7.70E-11**	**-2.34E-11**

Table 3.4-1
Risk Impact Analysis Results

System[1]	Safety Significance	Break Location	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank	SXI	RIS_B	Delta	w/POD	w/o POD	w/POD	w/o POD
SI	High	PLOCA	IGSCC	Medium	2	2	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SI	High	LOCA	(PWSCC)	Medium	1	0	-1	2.00E-12	2.00E-12	2.00E-13	2.00E-13
SI	High	LOCA	TASCS	Medium	2	8	6	-5.28E-10	-2.40E-10	-5.28E-11	-2.40E-11
SI	High	LOCA	None	Low	6	17	11	-2.20E-11	-2.20E-11	-2.20E-12	-2.20E-12
SI	High	PLOCA	None	Low	45	3	-42	2.10E-11	2.10E-11	2.10E-12	2.10E-12
SI	High	PPLOCA	None	Low	3	0	-3	1.50E-12	1.50E-12	1.50E-13	1.50E-13
SI	Low	Class 2 SI LSS		Assume Medium	70	0	-70	6.30E-09	6.30E-09	6.30E-10	6.30E-10
SI Total								**5.77E-09**	**6.06E-09**	**5.77E-10**	**6.06E-10**
CS Total	Low	Class 2 CS LSS		Assume Medium	12	0	-12	3.60E-09	3.60E-09	3.60E-10	3.60E-10
EF Total	Low	Class 2 EF LSS		Assume Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MS Total	Low	Class 2 LSS		Assume Medium	13	0	-13	1.30E-10	1.30E-10	1.30E-11	1.30E-11
SW Total	Low	Class 2 LSS		Assume Medium	4	0	-4	4.00E-11	4.00E-11	4.00E-12	4.00E-12
Grand Total					**217**	**79**	**-138**	**8.05E-09**	**9.29E-09**	**8.05E-10**	**9.29E-10**

(SLB = Steam Line Break)

Notes

1. Systems are described in Table 3.1.
2. The “Class 2 LSS” break location designation in Table 3.4-1 is used to identify those Code Class 2 locations that are not HSS because they do not meet any of the five HSS criteria of Section 2(a) of N-716 (e.g., not part of the BER scope).
3. The failure potential rank for high safety significant (HSS) locations is assigned as “High”, “Medium” or “Low” dependent upon potential susceptibly to the various types of degradation mechanisms. [Note: LSS locations were conservatively assumed to be a rank of Medium (i.e., “Assume Medium”).]
4. Only those ASME Section XI Code inspection locations that received a volumetric examination in addition to a surface examination are included in the count. Inspection locations previously subjected to a surface examination only were not considered in accordance with Section 3.7.1 of EPRI TR-112657.
5. Only those RIS_B inspection locations that receive a volumetric examination are included in the count. Locations subjected to VT2 only are not credited in count for risk impact assessment.

Table 5.1
Inspection Location Selections Comparison

System[1]	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N716	
	High	Low		DMs	Rank			Vol	Surface	RIS_B	Other
CH	x		LOCA	TT	Medium	B-F, B-J	32	2	10	10	NA
CH	x		ILOCA	TT	Medium	B-J	3	0	0	0	NA
CH	x		LOCA	None	Low	B-J	61	0	10	2	NA
CH	x		PLOCA	None	Low	B-J	7	0	0	0	NA
CH	x		ILOCA	None	Low	B-J	10	0	5	0	NA
CH		x	LSS	N/A	Assume Medium	C-F-1	5	0	0	0	NA
MFW	x		SLB	TASCS	Medium	C-F-2	6	3	0	6	NA
MFW	x		SLB	None	Low	C-F-2	42	4	0	1	NA
MFW	x		MFW-OC	None	Low	C-F-2	16	4	0	0	NA
RC	x		LOCA	PWSCC	Medium	B-J	1	0	1	0	NA
RC	x		LOCA	PWSCC,TT	Medium	B-J	1	1	0	0	NA
RC	x		LOCA	TASCS	Medium	B-J	9	0	0	3	NA
RC	x		LOCA	TASCS,TT	Medium	B-J	17	4	0	8	NA
RC	x		LOCA	TT	Medium	B-F, B-J	9	1	1	2	NA
RC	x		PLOCA	TT	Medium	B-J	6	2	1	2	NA
RC	x		LOCA	None	Low	B-F, B-J	225	38	34	15	2 VT-2
RC	x		ILOCA	None	Low	B-J	19	0	3	0	NA
RC	x	x	PLOCA	None	Low	B-J	29	0	7	0	NA
SI	x		PLOCA	IGSCC	Medium	B-J	5	2	0	2	NA
SI	x		LOCA	PWSCC	Medium	B-F	1	1	0	0	NA

Table 5.1
Inspection Location Selections Comparison

System[1]	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N716	
	High	Low		DMs	Rank			Vol	Surface	RIS_B	Other
SI	x		LOCA	TASCS	Medium	B-J	9	2	1	8	NA
SI	x		LOCA	None	Low	B-F, B-J	44	6	5	17	NA
SI	x		PLOCA	None	Low	B-J	204	45	9	3	NA
SI	x		PPLOCA	None	Low	C-F-1	33	3	0	0	NA
SI		x	SI LSS	N/A	Assume Medium	C-F-1	1272	70	25	0	NA
CS		x	CS LSS	N/A	Assume Medium	C-F-1	173	12	0	0	NA
EF		x	EF LSS	N/A	Assume Medium	C-F-2	94	0	0	0	NA
MS		x	LSS	N/A	Assume Medium	C-F-2	89	13	2	0	NA
SW		x	LSS	N/A	Assume Medium	C-F-2	24	4	0	0	NA

Notes

1. Systems are described in Table 3.1.
2. The failure potential rank for high safety significant (HSS) locations is assigned as “High”, “Medium” or “Low” dependent upon potential susceptibly to the various types of degradation mechanisms. [Note: LSS locations were conservatively assumed to be a rank of Medium (i.e., “Assume Medium”).]
3. The column labeled “Other” is generally used to identify plant augmented inspection program locations credited per Section 4 of Code Case N-716. Code Case N-716 allows the existing plant augmented inspection program for IGSCC (Categories B through G) in a BWR to be credited toward the 10% requirement. This option is not applicable for the ANO-2 RIS_B application. The “Other” column has been retained in this table solely for uniformity purposes with other RIS_B application template submittals and to indicate those selections that were VT-2 only per N-716; these were not credited in the risk impact assessment.

APPENDIX 1

GAP DISPOSITION

The ANO-2 Peer Review was held in July 2008. Each Fact & Observation (F&O) was entered into the Model Change Request (MCR) database. Table 1 lists all of the F&Os that potentially affect the PSA model. Level D MCRs (Level D is a grade assigned to those findings that are considered documentation only issues per Entergy Fleet Procedure EN-DC-151) and F&Os that are suggestions will not affect the PSA model and are not included.

Table 1 F&Os from RG 1.200 Peer Review (2008)

F&O Number	Level	MCR Number	Status
SY-A8-01, Finding	C	A2-3118	Closed
HR-D6-01, Finding	C	A2-3126	Closed
HR-G9-01, Finding	C	A2-3128	Closed
DA-A1a-01, Finding	C	A2-3129	Open
DA-C10-01, Finding	C	A2-3130	Open
IF-A1-01, Finding, Applicable to all internal flooding (IF) SRs.	B	A2-31332	Open
LE-D1b-01, Finding	C	A2-3144	Open
LE-D6-01, Finding	C	A2-3145	Open
LE-E4-01, Finding	C	A2-3147	Open
LE-F1b-01, Finding	C	A2-3148	Open

The items that are closed have been addressed in the ANO-2 Revision 4p02 Internal Events PSA Model.

All other F&Os, along with the disposition, are presented below.

1. DA-A1a-01 – MCR 3129

   <u>Issue</u>

   RG1.200 Peer Review F&O DA-A1a-01, Finding

   Boundary developed for emergency diesel generator (EDG) starting air was outlined in PRA-ES-01-003 included the start air system inside the component boundary. The CAFTA model had the starting air (SA) modeled with Basic Events (BEs) set greater than zero, effectively placing the starting air outside the component boundary. See F&O SY-A8-01 for details.

Disposition

The boundary for the EDG should include the starting air system. Resolution to this issue will be accomplished by setting the SA event probabilities to zero. However, resolution to this issue will not impact the CDF since the SA events are not significant enough to show up in the cutsets. Furthermore, the model is currently conservative with the SA system modeled outside the EDG boundary.

2. DA-C10-01 – MCR 3130

Issue

CAT I given based on information listed in Procedure PRA-A2-01-003S05 does not address decomposing the component failure mode into sub-elements (or causes) that are fully tested, then using tests that exercise specific sub-elements in their evaluation.

May be over-counting demands and run-hours for component boundaries that are not tested during evolution. Need to review component boundaries and tests counted in data collection to ensure that one sub-element does not have many more successes than another.

Update procedure CE-P-05.07 with process details that ensure the requirements described in CAT II/III are met.

Disposition

Resolution of this issue is not expected to result in a significant change to CDF, if any. The Maintenance Rule data that is used in the PSA model is assumed to be appropriate. If the MR data gathering is different than what is needed for the PSA model, the difference will be minimal and only slightly change the CDF results. Therefore, the impact to RI-ISI is considered negligible.

3. IF-A1-01 – MCR 31332

Issue

At the time of the peer review, the ANO2 IF analyses had not been completed to the point that it could be reviewed. Entergy intends to use the same IF methodology for all three of their PWRs with the Waterford-3 plant being the lead plant. The Waterford-3 IF analysis had been completed. Entergy requested that the peer review team review the IF methodology for Waterford to confirm that the methodology met the standard. Entergy needs to complete the ANO2 IF analyses using the Waterford-3 methodology. Entergy will need to specifically address dual unit issues for ANO1 and ANO2.

Disposition

The ANO-2 Internal Flooding Analysis has been completed and has been performed in accordance with the accepted methodology used at Waterford-3. This issue is considered to be resolved and no impact to RI ISI.

4. LE-D1b-01 – MCR 3144

Issue

There is no evidence of an evaluation of the impact of the accident progression conditions on containment seals, penetrations, etc. The model this is based on is related to NUREG/CR-6595 so consistency with NUREG/CR-6595 meets CC-I, but there is no discussion of the accident progression conditions on these elements.

Provide a discussion or assessment of the accident progression conditions on the containment conditions noted in the SR.

Disposition

This issue is primarily related to documentation. No change to the LERF results is expected in performing the proposed evaluation. Therefore, this issue will have no impact on the RI-ISI.

5. LE-D6-01 – MCR 3145

Issue

Containment isolation is addressed by top event (question) 3. This is based on a calc that is noted not to have been maintained up to date. Since it has not been maintained up-to-date, there is no confidence that the analysis represents a realistic assessment; therefore, this does not meet CC II.

The containment isolation calc needs to be updated or demonstrated (confirmed) to be up to date. This should include an assessment of the containment penetrations to provide an assessment of the total number of penetrations required to provide a realistic evaluation of containment isolation reliability.

Disposition

Due to the nature of the Containment Isolation system, not many changes to the physical plant are expected. Resolution of this issue will not affect CDF and is not expected to result in any significant change to LERF, if any. Therefore, the impact to RI ISI is considered negligible.

6. LE-E4-01 – MCR 3147

Issue

Although the majority of the SR requirements in these three top high level requirements are met, there is no indication that dependencies between multiple HFEs have been addressed.

The Level 1 assessment completed an evaluation of the dependencies between human actions in the model. A similar analysis should be completed for the human actions in the Level 2 analyses and between the Level 2 and Level 1 analyses to ensure all HEP dependencies are identified and addressed appropriately.

Disposition

Although not specifically addressed, a sensitivity analysis was performed in the most recent LERF analysis to study the effects of forming logical combinations of all post-core damage human actions. The results indicate that while LERF results are not insignificantly impacted by human actions, the overall impact on LERD is very small in relation to the percent of CDF. Therefore, the impact to RI ISI is considered to be insignificant

7. LE-F1b-01 – MCR 3148

Issue

There is no documented evidence that ANO2 compared their LERF results to the results of other similar plants to confirm the reasonableness of the results with respect to relative contribution and frequency and ranking of contributors.

Disposition

This issue is primarily related to documentation. No change to the LERF results is expected in performing the proposed comparison. Therefore, this issue will have no impact on the RI-ISI.

APPENDIX 2

PSA CONFIGURATION CONTROL

The below table indicates how the Entergy PSA configuration control meets the ASME Standard Section 1-5 (Configuration Control) Requirements (ASME/ANS RA-S-2008)

Section	How Entergy Meets this via Fleet Procedure EN-DC-151 (Section stated below) or other Procedure/Guide
1-5.2 PRA CONFIGURATION CONTROL PROGRAM A PRA Configuration Control Program shall be in place. It shall contain the following key elements:	EN-DC-151
(a) a process for monitoring PRA inputs and collecting new information	Sections 5.2 and 5.3 of EN-DC-151
(b) a process that maintains and upgrades the PRA to be consistent with the as-built, as operated plant	Sections 5.2, 5.3, 5.4 and 5.5 of EN-DC-151
(c) a process that ensures that the cumulative impact of pending changes is considered when applying the PRA	ENS Procedure CE-P-05.12 section 5.4.5
(d) a process that maintains configuration control of computer codes used to support PRA quantification	Entergy Fleet Procedure EN-IT-104 “Software Quality Assurance Program”
(e) documentation of the Program	Section 7.0 of EN-DC-151
1-5.3 MONITORING PRA INPUTS AND COLLECTING NEW INFORMATION The PRA Configuration Control Program shall include a process to monitor changes in the design, operation, maintenance, and industry-wide operational history that could affect the PRA.	• Section 5.2 of EN-DC-151 • EN-NE-G-006, Fleet Engineering Guide on Initiating Events Analysis for PSA • EN-NE-G-007, Fleet Engineering Guide on Data Analysis for PSA
These changes shall include inputs that impact operating procedures, design configuration, initiating event frequencies, system or subsystem unavailability, and component failure rates.	• Section 5.2 of EN-DC-151 • EN-NE-G-006, Fleet Engineering Guide on Initiating Events Analysis for PSA • EN-NE-G-007, Fleet Engineering Guide on Data Analysis for PSA

Section	How Entergy Meets this via Fleet Procedure EN-DC-151 (Section stated below) or other Procedure/Guide
The program should include monitoring of changes to the PRA technology and industry experience that could change the results of the PRA model.	Section 5.5 sub-section [5] of EN-DC-151
1-5.4 PRA MAINTENANCE AND UPGRADES The PRA shall be maintained and upgraded, such that its representation of the as-built, as-operated plant is sufficient to support the applications for which it is being used.	Section 5.6 [1] of EN-DC-151
Changes in PRA inputs or discovery of new information identified pursuant to 1-5.3 shall be evaluated to determine whether such information warrants PRA maintenance or PRA upgrade. (See Section 1-2 for the distinction between PRA maintenance and PRA upgrade.)	Section 5.3 [1] of EN-DC-151
Changes that would impact risk-informed decisions should be incorporated as soon as practical.	Sections 5.4 and 5.5 of EN-DC-151
Changes that are relevant to a specific application shall meet the SRs pertinent to that application as determined through the process described in 1-3.5.	ENS Procedure CE-P-05.12
Changes to a PRA due to PRA maintenance and PRA upgrade shall meet the requirements of the Technical Requirements Section of each respective Part of this Standard.	This is fully the intent of the PSA guidelines and fleet procedure EN-DC-151. It should be noted that the PSA guidelines have been recently issued, and a few guidelines are in process in 2010. Entergy is transitioning to this process with the new major PSA updates.

Section	How Entergy Meets this via Fleet Procedure EN-DC-151 (Section stated below) or other Procedure/Guide
Upgrades of a PRA shall receive a peer review in accordance with the requirements specified in the Peer Review Section of each respective Part of this Standard, but limited to aspects of the PRA that have been upgraded.	Section 5.5 [7] of EN-DC-151
1-5.5 PENDING CHANGES This Standard recognizes that immediately following a plant change [e.g., modifications, procedure changes, plant performance (data)], or upon identification of a subject for model improvement (e.g., new human error analysis methodology, new data update methods), a PRA may not represent the plant until the subject plant change or model improvement is incorporated into the PRA. Therefore, the PRA configuration control process shall consider the cumulative impact of pending plant changes or model improvements on the application being performed. The impact of these plant changes or model improvements on the results of the PRA and the decision under consideration in the application shall be evaluated in a fashion similar to the approach used in Section 1-3.	ENS Procedure CE-P-05.12
1-5.6 USE OF COMPUTER CODES The computer codes used to support and to perform PRA analyses shall be controlled to ensure consistent, reproducible results.	Entergy Fleet Procedure EN-IT-104 “Software Quality Assurance Program”

Section	How Entergy Meets this via Fleet Procedure EN-DC-151 (Section stated below) or other Procedure/Guide
1-5.7 DOCUMENTATION Documentation of the Configuration Control Program and of the performance of the above elements shall be adequate to demonstrate that the PRA is being maintained consistently with the as-built, as-operated plant. The documentation typically includes	
(a) a description of the process used to monitor PRA inputs and collect new information	Section 5.2 of EN-DC-151and MCR Database
(b) evidence that the aforementioned process is active	Though Entergy possess no formal document on activity or frequency, Entergy's MCR process for PSA issues is a very active one. MCRs are being written frequently as issues are identified. For example, almost 300 MCRs were written on the 5 southern PSA models in a recent 6 month period.
(c) descriptions of proposed changes	The MCR Database has this information.
(d) description of changes in a PRA due to each PRA upgrade or PRA maintenance	PRA Summary Report prepared following a periodic update
(e) record of the performance and results of the appropriate PRA reviews (consistent with the requirements of 1-6.6)	Signoff of calculations
(f) record of the process and results used to address the cumulative impact of pending changes	MCR Database and CE-P-05.12
(g) a description of the process used to maintain software configuration control	Software qualification packages per EN-IT-104

Attachment 2 to

2CAN011001

List of Regulatory Commitments

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
ANO-1 is in the process of evaluating MRP-146, *Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines*, and these results will be incorporated into the RIS_B Program, as warranted.	✓		June 30, 2011
The request for alternative pertaining to the use of Code Case N-578 will be withdrawn for use at ANO-1 upon NRC approval of the RIS_B Program submittal.	✓		Upon NRC approval of this request for alternative
Upon approval of the RIS_B Program, procedures that comply with the guidelines described in EPRI TR-112657 will be prepared to implement and monitor the program.	✓		Upon NRC approval of this request for alternative