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1CAN060902

June 11, 2009

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 1
Docket No. 50-313
License No. DPR-51

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 based on the American Society of Mechanical Engineers (ASME) Code Case N-716, as documented in the attached Request for Alternative ANO1-ISI-014. ANO1-ISI-014 is being submitted in a template format in Attachment 1. This template format is similar to the submittals the NRC Staff has approved for Waterford 3 and Grand Gulf. This format is also similar to the recently submitted request for alternative by Calvert Cliffs for the same subject.

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.

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

Entergy requests approval of the proposed alternative by June 1, 2010. ANO-1 will withdraw the Request for Alternative CEP-ISI-007 pertaining to the application of Code Case N-663 for use at ANO-1 upon NRC approval of this RI ISI program submittal. 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

Attachments:

1. Request for Alternative ANO1-ISI-014
2. List of Regulatory Commitments

cc: Mr. Elmo E. Collins
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Attachment 1 to

1CAN06902

Request for Alternative

ANO1-ISI-014

REQUEST FOR ALTERNATIVE
ENTERGY OPERATIONS, INC.
ARKANSAS NUCLEAR ONE – UNIT 1

REQUEST FOR ALTERNATIVE
ANO1-ISI-014

APPLICATION OF ASME CODE CASE N-716

RISK-INFORMED / SAFETY-BASED INSERVICE INSPECTION PROGRAM PLAN

Table of Contents

ENTERGY OPERATIONS, INC.
ARKANSAS NUCLEAR ONE – UNIT 1

REQUEST FOR ALTERNATIVE
ANO1-ISI-014

1. INTRODUCTION

Arkansas Nuclear One – Unit 1 (ANO-1) is currently in the fourth 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-1 plans to implement a risk-informed / safety-based inservice inspection (RIS_B) program during the first inspection period of this interval.

The ASME Section XI code of record for the fourth ISI interval at ANO-1 is 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-1 Level 1 PSA was initially developed in response to the NRC Generic Letter (GL) 88-20 on Individual Plant Examinations (IPEs). The ANO-1 IPE model was developed by ANO Design Engineering staff, Science Applications International Corporation, and ERIN Engineering and Research, Inc. The IPE was submitted to the NRC in April 1992. The ANO-1 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-1 IPE results. The letter concluded that the ANO-1 IPE met the intent of GL 88-20; that is, the ANO-1 IPE process was capable of identifying the most likely severe accidents and severe accident vulnerabilities for ANO-1.

As part of the IPE development process, an expert panel review was performed on the results. This panel was composed of experienced personnel selected by the Babcock and Wilcox Owners Group. The comments from the IPE independent review were addressed, prior to its March 1992 submittal to NRC.

Several PSA model updates have been completed on the ANO-1 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-1 PSA is more than adequate for this application. The PSA model used for this application is based on a gap evaluation to RG 1.200, Revision 1 and the gaps have been evaluated. Most gaps have been resolved and incorporated into the model used for this application. Several others are documentation only concerns and the remaining were reviewed relative their potential impact on this application. Entergy believes the ANO-1 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.

Industry Peer Review in 2002

An industry peer review of the ANO-1 PSA was conducted in 2002 on the Revision 2p2 of the PSA model. The peer review concluded that there were several areas where the ANO-1 model needed improvement. The ANO-1 PSA model update (Rev 3p0) completed in August 2006 addressed most of the significant Facts and Observations (F&O's) from this peer review.

3p0 Major Model Update

The ANO-1 Revision 3p0 Level-1 PRA Model Update was a major update with substantial changes to most of the model components. New PSA software and utilities were used for Revision 3p0 update. A list of major changes from ANO-1 PSA Model 2p2 to PSA Model 3p0 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 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 In Revision 3p0:
  - Anticipated Trip Without a Scram (ATWS): New ATWS event trees, developed the Reactor Protection System (RPS) / Diverse Reactor Overpressure Protection System (DROPS) / 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 Injection (HPI)
  - Low Pressure Injection / Decay Heat Removal (LPI / DHR)
  - Service Water
  - Emergency Feedwater (EFW)
  - AC/DC
  - Instrument Air

- Other Significant Model Components Updated:
  - Accident Sequences / Top Logic
  - Plant-Specific Data
  - Reliability Database
  - Closed more than 200 ANO-1 Model Change Requests (MCRs) against Revision 2p2

The Rev 3p0 baseline total Core Damage Frequency (CDF) was 2.419E-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. In preparation for ANO-1's transition to National Fire Protection Association (NFPA) 805 standard, a gap assessment of the ANO-1 PSA 3p0 model has been completed. Gaps to the ASME PSA standard and RG 1.200, Revision 1 have been identified. The gaps impacting the fire PRA were closed in the near term in order to meet the NFPA transition schedule.

Model Revision 4p0

The ANO-1 Internal Events PSA model is currently undergoing a major model update (4p0) to meet the RG 1.200 Revision 1 standards. It is expected that the significant model gaps to the ASME Standard impacting the fire PRA will be closed with the Revision 4 Model Update. The version used in the subject evaluation is an interim model as the 4p0 update is still in progress, slated to be complete in Summer 2009. The interim model used in this RI ISI evaluation has a number of RG 1.200 gaps already closed. The remaining open gaps are dispositioned for the RI ISI application as detailed in Appendix 1. The baseline CDF for this model is

4.650E-06/rx-yr. Though model update calculations have not been signed off for this interim model, each model change was made through the Model Change Request (MCR) database process. Each MCR that was addressed has been reviewed by an independent person and approved by the PSA supervisor. Also, a cutset review by the PSA Group has been conducted on the top cutsets from this interim model. Issues identified during this cutset review were corrected before the model was used in the RI ISI evaluation. As the results from this interim model have undergone this review, Entergy has a high level of confidence that this model is acceptable for this particular application. This acceptability is further confirmed by the very low values of CDF and LERF calculated in the internal flooding calculation for all of the scenarios; these values are well below the thresholds in the Code Case N-716.

Internal Flooding Model

The ANO-1 Internal Flooding Analysis (IFA) was significantly upgraded to meet the requirements of RG 1.200, Revision 1 in 2008. 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 some 60-odd quantified scenarios; with CDF ranging from the 1E-8 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-1 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 interim Rev 4p0 Internal Events CDF model to generate the LERF for the IFA scenarios. The baseline LERF for this model is 9.83E-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 RI ISI program based on ASME Code Case N-560 was approved for use at ANO-1 by the NRC on August 25, 1999. This RI ISI Program only applies to Class 1 Category B-J piping welds (excluding socket welds).

- ASME Code Case N-663 (Request for Alternative CEP-ISI-007) was approved for use at ANO-1 by the NRC on August 26, 2003. For ANO-1, this alternative only applies to Class 2 Category C-F-1 and C-F-2 piping welds.

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 Arkansas Nuclear One Upper Level Document ULD-0-TOP-07, *HELB/MELB Topical ULD* and Calculation 86D-1005-29, 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, this plant augmented inspection programs are subsumed by the new RIS_B Program.

- The plant augmented inspection program for flow accelerated corrosion (FAC) per 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-1 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-1 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 Pure 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-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, if 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 shutdown cooling pressure boundary function. That is, Class 1 and 2 welds of systems or portions of systems needed to utilize the normal shutdown cooling 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-1. 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

   $\Delta T > 50^{o}F$,

   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 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-1 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-1 RIS_B application, if 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 Reactor Pressure Vessel (RPV)) and the RPV.

(4) A minimum of 10% of the welds in that portion of the RCPB that lies outside containment (OC) (e.g., portions of the main feedwater system in BWRs) 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
1	417	44	1661	17	6	2	2084	63

Notes

(1) Includes all Category B-F and B-J locations. All 417 Class 1 piping weld locations are HSS.

(2) Includes all Category C-F-1 and C-F-2 locations. Of the 1661 Class 2 piping weld locations, 191 are HSS and the remaining 1470 are LSS.

(3) All six of these non-nuclear safety (NNS) piping weld locations are 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 is dependent on practical considerations (i.e., the practicality of performing additional 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-1 will calculate coverage and use additional examinations or techniques in the same manner it has 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 examination of Class 1 Category B-J piping welds (excluding socket welds) based on the use of Code Case N-560, Request for Alternative CEP-ISI-007 pertaining to the application of Code Case N-663 and Request for Alternative 96-003 pertaining to the performance of automated ultrasonic examinations from the pipe inside diameter (ID) in lieu of surface examinations from the pipe outside diameter (OD) for the reactor vessel core flood safe end-to-nozzle piping weld locations will be withdrawn for use at ANO-1 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, and 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-1 RIS_B application, CCDP and CLERP values of 2.8E-4 and 5.6E-5 have been used for LSS welds to bound plant internal flooding study results. The 2.8E-4 and 5.6E-5 values used for CCDP and CLERP was determined based on results from the plant internal flooding study for HPI piping and have been conservatively applied as an upper bound for all LSS welds.

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 done 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 (“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-1 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 break locations that fall within the high consequence rank range was based on the highest CCDP value obtained (i.e., ANO-1 plant-specific flood scenario AB356-77 that contains portions of the main feedwater system piping lines outside of containment).

CCDP and CLERP Values Based on Break Location

Break Location Designation	Estimated		Consequence Rank	Upper Bound	
	CCDP	CLERP		CCDP	CLERP
LOCA	1.57E-03	3.14E-04	**HIGH**	1.65E-03	3.30E-04
RCPB pipe breaks that result in a LOCA – Based on highest CCDP for LOCA (Large Break) from PSA model (0.2 margin used for CLERP)					
PLOCA	4.07E-07	8.14E-08	**MEDIUM**(1)	1.00E-04	2.00E-05
RCPB pipe breaks that result in a potential LOCA – Based on Large LOCA CCDP of 1.57E-3 and valve rupture probability of 2.59E-4 (0.2 margin used for CLERP)					
PLOCA – SD	1.26E-06	2.52E-07	**MEDIUM**	1.00E-04	2.00E-05
RCPB pipe breaks that occur in shutdown cooling piping and result in a potential LOCA – Based on analysis of RCS Draindown event (>1000 gallons) since any size LOCA has such a low probability of occurring during shutdown conditions (0.2 margin used for CLERP)					
SD	1.26E-06	2.52E-07	**MEDIUM**	1.00E-04	2.00E-05
Class 2 pipe breaks that occur in shutdown cooling piping inside containment – Based on analysis of RCS Draindown event (>1000 gallons) using the WF3 SD model which is very similar to the ANO-1 SD model but quantifiable (0.2 margin used for CLERP)					
MFW – 1, MS – 2 and 3	6.3E-07	1.26E-07	**MEDIUM**	1.00E-04	2.00E-05
Class 2 pipe breaks that occur in main feedwater piping inside containment [1], main steam piping outside containment between containment penetration and main steam isolation valve [2] and main steam piping outside containment downstream of main steam isolation valve [3] – Based on a steamline/feedline break (0.2 margin used for CLERP)					
MFW – 2, 3, and 4	1.65E-03	3.30E-04	**HIGH**	1.65E-03	3.30E-04
Class 2 pipe breaks that occur in main feedwater piping outside containment between check valve and containment penetration [2], main feedwater piping outside containment between main feedwater isolation valve and check valve [3] and main feedwater piping outside containment upstream of main feedwater isolation valve [4] – Based on ANO-1 plant specific flood scenario AB356-77 that contains portions of the feedlines outside of containment (0.2 margin used for CLERP)					
Class 2 LSS	2.80E-04	5.60E-05	**HIGH**	2.80E-04	5.60E-05
Class 2 pipe breaks that occur in the remaining system piping designated as LSS – Based on ANO-1 plant specific flood scenario AB335-53-A that contains Class 2 LSS HPI piping (0.2 margin used for CLERP)					

Note

1. 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.

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 1992 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 are 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.

ANO-1 Risk Impact Results

System[(1)]	ΔR_{CDF} Results		ΔR_{LERF} Results	
	w/ POD	w/o POD	w/ POD	w/o POD
RC	-2.25E-09	-2.06E-10	-4.50E-10	-4.13E-11
MUP	-1.86E-09	9.75E-10	-3.73E-10	1.95E-10
DH	4.37E-10	9.65E-10	8.74E-11	1.93E-10
MFW	-1.45E-11	-1.45E-11	-2.90E-12	-2.90E-12
MS	5.01E-10	5.17E-10	1.00E-10	1.03E-10
EFW	2.80E-11	2.80E-11	5.60E-12	5.60E-12
RBS	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL	-3.16E-09	2.26E-09	-6.33E-10	4.53E-10

Note

1. Systems are described in Table 3.1.

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 leak 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-1 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 into 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 GL requirements, or by industry and plant-specific feedback.

For preservice examinations, ANO-1 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. 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 1992 Code Edition program requirements for in-scope piping is provided in Table 5.

ANO-1 intends to start implementing the RIS_B Program during the plant's first period of the current (fourth) inspection interval. The fourth ISI interval will implement 100% of the inspection locations selected for examination per the RIS_B Program. Examinations shall be performed such that the period percentage requirements of ASME Section XI are met.

6. REFERENCES/DOCUMENTATION

USNRC Safety Evaluation pertaining to the use of ASME Code Case N-560, dated August 25, 1999 (Letter 1CNA089904)

USNRC Safety Evaluation pertaining to the use of ASME Code Case N-663, dated August 26, 2003 (Letter CNRI-2003-00010)

USNRC Safety Evaluation pertaining to the performance of automated ultrasonic examinations from the pipe ID in lieu of surface examinations from the pipe OD for the reactor vessel core flood safe end-to-nozzle piping weld locations, dated February 23, 2003 (Letter 1CNA029906)

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

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

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

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

Supporting Onsite Documentation

ENTP-19Q-320, *Degradation Mechanism Evaluation for ANO-1*, Revision 2

ENTP-19Q-321, *N-716 Evaluation of Arkansas Nuclear One – Unit 1*, Revision 1

APPENDIX 1

GAP DISPOSITION

The ANO-1 Gap Analysis was held in June 2007. Each Fact and Observations (F&Os) were entered into the Model Change Request (MCR) database. Table 1 lists all of the F&Os identified along with the MCR associated with that F&O and the MCR status.

Table 1 ALL F&Os and Associated MCR

F&O Number	Level	MCR Number	Status
IE-GA-0001	B	A1-2727	Closed
IE-GA-0002	A	A1-2729	Closed
IE-GA-0003	B	A1-2730	Closed
IE-GA-0004	C	A1-2731	Closed
IE-GA-0005	C	A1-2732	Closed
IE-GA-0006	A	A1-2733	Closed
IE-GA-0007	B	A1-2734	Open
IE-GA-0008	B	A1-2737	Closed
IE-GA-0009	D	A1-2738	Closed
IE-GA-0010	A	A1-2739	Closed
IE-GA-0011	D	A1-2740	Closed
IE-GA-0012	B	A1-2741	Closed
IE-GA-0013	A	A1-2742	Closed
IE-GA-0014	A	A1-2743	Closed
IE-GA-0015	A	A1-2744	Closed
IE-GA-0016	B	A1-2745	Closed
AS-GA-0001	B	A1-2746	Closed
AS-GA-0002	B	A1-2747	Closed

Table 1
ALL F&Os and Associated MCR
(continued)

F&O Number	Level	MCR Number	Status
AS-GA-0003	B	A1-2750	Closed
AS-GA-0004	B	A1-2748	Closed
AS-GA-0005	A	A1-2749	Closed
AS-GA-0006	C	A1-2751	Closed
AS-GA-0007	B	A1-2752	Closed
AS-GA-0008	B	A1-2753	Closed
AS-GA-0009	B	A1-2754	Closed
AS-GA-0010	B	A1-2755	Closed
SC-GA-0001	B	A1-2756	Closed
SC-GA-0002	B	A1-2757	Closed
SC-GA-0003	B	A1-2758	Closed
SC-GA-0004	B	A1-2759	Closed
SC-GA-0005	B	A1-2760	Closed
SY-GA-0001	A	A1-2761	Closed
SY-GA-0002	B	A1-2762	Open
SY-GA-0003	A	A1-2763	Closed
SY-GA-0004	B	A1-2764	Open
SY-GA-0005	A	A1-2765	Closed
SY-GA-0006	A	A1-2766	Closed
SY-GA-0007	A	A1-2767	Closed
SY-GA-0008	A	A1-2768	Closed
SY-GA-0009	B	A1-2769	Open

Table 1
ALL F&Os and Associated MCR
(continued)

F&O Number	Level	MCR Number	Status
SY-GA-0010	B	A1-2770	Open
SY-GA-0011	B	A1-2771	Closed
SY-GA-0012	A	A1-2772	Closed
SY-GA-0013	B	A1-2773	Closed
SY-GA-0014	B	A1-2774	Closed
SY-GA-0015	B	A1-2775	Closed
SY-GA-0016	B	A1-2776	Open
SY-GA-0017	B	A1-2777	Open
SY-GA-0018	B	A1-2778	Open
SY-GA-0019	B	A1-2779	Closed
SY-GA-0020	B	A1-2780	Open
HR-GA-0001	C	A1-2781	Closed
HR-GA-0002	C	A1-2782	Closed
HR-GA-0003	B	A1-2783	Closed
HR-GA-0004	C	A1-2784	Closed
HR-GA-0005	A	A1-2785	Closed
HR-GA-0006	B	A1-2786	Closed
HR-GA-0007	C	A1-2787	Closed
HR-GA-0008	C	A1-2788	Closed
HR-GA-0009	B	A1-2789	Closed
DA-GA-0001	A	A1-2790	Closed
DA-GA-0002	B	A1-2791	Closed

Table 1
ALL F&Os and Associated MCR
(continued)

F&O Number	Level	MCR Number	Status
DA-GA-0003	A	A1-2792	Open
DA-GA-0004	A	A1-2793	Closed
DA-GA-0005	B	A1-2794	Closed
DA-GA-0006	B	A1-2795	Open
DA-GA-0007	B	A1-2796	Open
DA-GA-0008	B	A1-2797	Closed
DA-GA-0009	B	A1-2798	Open
DA-GA-0010	B	A1-2799	Open
DA-GA-0011	B	A1-2800	Open
DA-GA-0012	B	A1-2801	Open
DA-GA-0013	C	A1-2802	Open
DA-GA-0014	B	A1-2803	Open
QU-GA-0001	B	A1-2804	Open
QU-GA-0002	B	A1-2805	Closed
QU-GA-0003	C	A1-2806	Open
QU-GA-0004	D	A1-2807	Closed
QU-GA-0005	B	A1-2808	Closed
QU-GA-0006	B	A1-2809	Open
QU-GA-0007	B	A1-2810	Open

The items that are closed out have been addressed in the ANO-1 Revision 4 Internal Events PSA Model (R4 Model) used for this application. Those items that address documentation issues only and have no impact on the RI ISI submittal are summarized in Table 2. These items will be addressed during the Revision 4 update prior to the scheduled peer review in August 2009.

Table 2
Open F&Os related to Documentation Only

F&O Number	Level	MCR Numbers
IE-GA-0007	B	A1-2734
SY-GA-0002	B	A1-2762
SY-GA-0004	B	A1-2764
SY-GA-0009	B	A1-2769
SY-GA-0016	B	A1-2776
SY-GA-0017	B	A1-2777
SY-GA-0020	B	A1-2780
DA-GA-0006	B	A1-2795
DA-GA-0007	B	A1-2796
DA-GA-0009	B	A1-2798
DA-GA-0012	B	A1-2801
DA-GA-0014	B	A1-2803
QU-GA-0003	C	A1-2806
QU-GA-0007	B	A1-2810

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

UNRESOLVED F&Os

1 F&O -- SY-GA-0010, Level B Finding – MCR A1-2770

1.1 <u>Issue</u>

There is no documentation regarding searches or identified events related to concurrent maintenance as required by the ASME standard. Documentation related to searches through historical data and/or maintenance procedures are needed.

1.2 Disposition

This issue involves documentation of concurrent maintenance searches. It may require a review of plant maintenance experience or interviews with plant staff. At a minimum, it requires documentation of reviews already performed.

This item is not expected to affect the results to be applied to the RI ISI application.

2 F&O -- SY-GA-0018, Level B Finding – MCR A1-2778

2.1 Issue

An assessment of design basis EQ in the system notebook is needed to address the requirements in the ASME standard.

2.2 Disposition

This is mainly a documentation issue, may require some information gathering. This issue is not expected to affect the results to be applied to the RI ISI application.

3 F&O -- DA-GA-0003, Level A Finding – MCR A1-2792

3.1 Issue

The CAFTA type code contains unique failure type codes for similar components in different systems. However, the data values are the same (for example the failure to open mode for motor-operated valves). It appears that no separation of data is embodied in the source data. Using different type codes for the same data may preclude performing sensitivity studies on data failure rates and collecting uncertainty/sensitivity information. Separate type codes should only be applied where unique data is employed such that event data from the same source can be addressed.

3.2 Disposition

Changing the type code impacts the parametric uncertainty by making the UNCERT distribution narrower. It would have at most a very minor impact on the baseline mean risk values reported or any information used in the RI ISI. The impact on the risk conclusions is expected to be very insignificant.

4 F&O -- DA-GA-0010, Level B Finding – MCR A1-2799

4.1 Issue

Depending on the calculated value, this could be elevated to an "A" finding. There is currently no method to determine the times at which coincident maintenance was occurring since only a total is provided. This ASME requirement states that for a Category II study the analysis must be able to define and calculate this type of unavailability for both inter- and intra-system cases.

4.2 Disposition

This issue could possibly be addressed by updating documentation which may require some information gathering. If coincident maintenance events are found that are not included in the PSA, they can be addressed by a sensitivity case, adding these coincident maintenance unavailability events to the model. This is not expected to materially impact the results with respect to the RI ISI application.

5 F&O -- DA-GA-0011, Level B Finding – MCR A1-2800

5.1 Issue

Depending on the calculated values for additional components, this could be an "A" finding. The current assessment does address most major components with regard to plant-specific data. However, some of the typically addressed components with data typically available, i.e., motor-operated valves, are not included and are not excluded based on a lack of data. The current assessment should be expanded to include more components where data is available or documentation provided that supports their exclusion.

5.2 Disposition

The components that were explicitly pointed out in this finding are valves. From previous experience, it has been demonstrated that when a valve has a failure, the plant very quickly changes or replaces the valve or its failed part. Entergy believes that the margins in the internal flooding CDF and LERF results to the thresholds in Code Case N-716 are well above the impact of using plant specific failure data that might possibly be outliers compared to the generic failure data. Recent industry generic failure rates have shown a decreasing trend as reported in NUREG/CR-6928, and ANO is not believed to be an outlier in this respect. Therefore, this issue will have no material effect on the RI ISI application

6 F&O -- DA-GA-0013, Level C Finding – MCR A1-2802

6.1 Issue

IEEE-500-1984 was used for some mechanical components. This is considered to be a data source of last resort. ANO-1 should try to find a better generic data source for these components. Echelon Calculation PRA-ES-01-001, Revision 1 does state that IEEE-500 was used for six component failure modes because IEEE-500 was the only source of data that could be found for the specific component failure modes.

6.2 Disposition

This is mainly a documentation issue. Values used from IEEE-500 for mechanical components are higher than corresponding values from NUREG/CR-6928 (latest generic data).

PRA-ES-01-001 has been updated by Echelon Calculation PRA-ES-01-003. PRA-ES-01-003 uses updated industry guidance from NUREG/CR-6928. PRA models maintained by Echelon now only have three failure modes and two type codes issued from IEEE-500.

Failure Modes:

Control relay spurious operation 6.0 E-08 per hour

DC disconnect switch transfers open 1.4 E-06 per hour

Nozzle failure due to plugging 1.80E-05 per hour

Associated Type Codes:

AST HI Temperature Switch, Fails High 1.00E-06 N

SPC DN Electronic Control Power Supply Does Not Operate 1.60E-06 H

No updates for these specific modes were available. IEEE Standard 500-1984 has been withdrawn, and is no longer endorsed by the IEEE. However, even though its data may be outdated or obsolete, IEEE Standard 500-1984 provides the only known source of data for certain components and has therefore been used in PRA-ES-01-001 (which is a reference for PRA-ES-01-003).

7 F&O -- QU-GA-0001, Level B Finding – MCR A1-2804

7.1 Issue

For the "state-of-knowledge" correlation, the basic event probabilities use the same failure probability. However, the ANO-1 model breaks these type codes by system and thus impacts the correlation.

7.2 Disposition

Changing the type code impacts the parametric uncertainty by making the UNCERT distribution narrower. As described in F&O 3 above, it would have at most a minor impact on the baseline mean risk values reported or any information used in the R-I ISI.

8 F&O -- QU-GA-0006, Level B Finding – MCR A1-2809

8.1 Issue

The summary report does not include any discussion of the model limitations that would impact the quantification process. Among the limitations in the model are the lack of internal flooding or external events and the change in importances due to system alignments.

8.2 Disposition

Internal flooding has been updated to conform to the RG 1.200, Revision 1 requirements. Discussion of changes in importances due to system alignments does not appreciably impact the results used in the RI ISI submittal. Documentation will be updated during the Integration & Quantification update which is currently in progress.

APPENDIX 2

PSA CONFIGURATION CONTROL

The below table indicates how the Entergy PSA configuration control meets ASME/ANS RA-Sa-2009, Addenda to ASME/ANS RA-S-2008, Standard for Level 1 / large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, 2009.

Section	**How Entergy Meets this via Fleet Procedure EN-DC-151 (Section stated below) or other Procedure/Guide**
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
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 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

Section	How Entergy Meets this via Fleet Procedure EN-DC-151 (Section stated below) or other Procedure/Guide
(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 IT 104 "Software Quality Assurance"
(e) documentation of the Program	Section 7.0 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 are in process in the next several months. 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 IT 104 “Software Quality Assurance”

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 the last 5 months.
(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 – each calculation
(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 IT-104

Table 3.1
N-716 Safety Significance Determination

System Description	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	>1E-6 CDF >1E-7 LERF	High	Low
RC – Reactor Coolant	2	✓	✓				✓	
	186	✓					✓	
MUP – Makeup and Purification	175	✓					✓	
	705							✓
DH – Decay Heat Removal	54	✓	✓				✓	
	90		✓				✓	
	451							✓
MFW – Main Feedwater	97			✓			✓	
	2				✓		✓	
MS – Main Steam	8				✓		✓	
	130							✓
EFW – Emergency Feedwater	8							✓
RBS – Reactor Building Spray	176							✓
SUMMARY RESULTS FOR ALL SYSTEMS	56	✓	✓				✓	
	361	✓					✓	
	90		✓				✓	
	97			✓			✓	
	10				✓		✓	
	1470							✓
TOTALS	**2084**						**614**	**1470**

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
RC	✓	✓				✓					
MUP[2]		✓	✓								
DH[2]		✓				✓					
MFW											✓
MS[2]		✓									
EFW[2]											
RBS[2]											

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 EFW and RBS systems in their entirety, as well as portions of the MUP, DH and MS systems.

Table 3.3
N-716 Element Selections

System[1]	Selections	HSS	DMs[2]		RCPB	RCPB^{IFIV}	RCPB^{OC}	BER
RC	**Required**	19 of 188	TASCS, TT, (PWSCC)	1 of 3	19 of 188	13	n/a	n/a
			TASCS, TT	7 of 25				
			TT, (PWSCC)	1 of 3				
			TT	1 of 3				
			None (PWSCC)	1 of 3				
	Made	19	TASCS, TT, (PWSCC)	2	19	19	n/a	n/a
			TASCS, TT	7				
			TT, (PWSCC)	1				
			TT	1				
			None (PWSCC)	1				
MU	**Required**	18 of 175	TT, IGSCC		18 of 175	12	n/a	n/a
			TT	18(25) of 100				
			IGSCC					
	Made	18	TT, IGSCC	1	18	16	n/a	n/a
			TT	16				
			IGSCC	1				
DH	**Required**	15 of 144	TT	2 of 7	6 of 51	4	n/a	n/a
			None (PWSCC)	1 of 1				
	Made	15	TT	2	7	7	n/a	n/a
			None (PWSCC)	1				
MFW	**Required**	10 of 99	n/a		n/a	n/a	n/a	1 of 2
	Made	10	n/a		n/a	n/a	n/a	2
MS	**Required**	1 of 8	TT	1 of 2	n/a	n/a	n/a	1 of 8
	Made	1	TT	1	n/a	n/a	n/a	1
EFW	**Required**	n/a	n/a		n/a	n/a	n/a	n/a
	Made	n/a	n/a		n/a	n/a	n/a	n/a
RBS	**Required**	n/a	n/a		n/a	n/a	n/a	n/a
	Made	n/a	n/a		n/a	n/a	n/a	n/a
TOTAL	**Made**	**63**	**34**		**44**	**42**	**n/a**	**3**

Note

1. Systems are described in Table 3.1.
2. For the makeup and purification system, no more than 10% of the HSS piping welds are required to be selected for examination.

Table 3.4-1
Risk Impact Analysis Results

System[(1)]	Safety Significance	Break Location[(2)]	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank[(3)]	SXI[(4)]	RIS_B	Delta	w/ POD	w/o POD	w/ POD	w/o POD
RC	High	LOCA	TASCS, TT, (PWSCC)	Medium (Medium)	1	2	1	-4.95E-10	-1.65E-10	-9.90E-11	-3.30E-11
RC	High	LOCA	TASCS, TT	Medium	5	7	2	-1.58E-09	-3.30E-10	-3.17E-10	-6.60E-11
RC	High	LOCA	TT, (PWSCC)	Medium (Medium)	3	1	-2	0.00E+00	3.30E-10	0.00E+00	6.60E-11
RC	High	LOCA	TT	Medium	0	1	1	-2.97E-10	-1.65E-10	-5.94E-11	-3.30E-11
RC	High	LOCA	None (PWSCC)	Low (Medium)	3	1	-2	1.65E-11	1.65E-11	3.30E-12	3.30E-12
RC	High	LOCA	None	Low	20	7	-13	1.07E-10	1.07E-10	2.15E-11	2.15E-11
RC	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL								**-2.25E-09**	**-2.06E-10**	**-4.50E-10**	**-4.13E-11**
MUP	High	PLOCA	TT, IGSCC	Medium	0	1	1	-1.00E-11	-1.00E-11	-2.00E-12	-2.00E-12
MUP	High	LOCA	TT	Medium	11	16	5	-3.66E-09	-8.25E-10	-7.33E-10	-1.65E-10
MUP	High	PLOCA	IGSCC	Medium	0	1	1	-1.00E-11	-1.00E-11	-2.00E-12	-2.00E-12
MUP	High	LOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MUP	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MUP	Low	Class 2 LSS	N/A	Assume Medium	65	0	-65	1.82E-09	1.82E-09	3.64E-10	3.64E-10
TOTAL								**-1.86E-09**	**9.75E-10**	**-3.73E-10**	**1.95E-10**
DH	High	LOCA	TT	Medium	4	2	-2	-1.98E-10	3.30E-10	-3.96E-11	6.60E-11
DH	High	LOCA	None (PWSCC)	Low (Medium)	0	1	1	-8.25E-12	-8.25E-12	-1.65E-12	-1.65E-12
DH	High	LOCA	None	Low	6	4	-2	1.65E-11	1.65E-11	3.30E-12	3.30E-12
DH	High	PLOCA – SD	None	Low	4	0	-4	2.00E-12	2.00E-12	4.00E-13	4.00E-13
DH	High	SD	None	Low	25	8	-17	8.50E-12	8.50E-12	1.70E-12	1.70E-12
DH	Low	Class 2 LSS	N/A	Assume Medium	22	0	-22	6.16E-10	6.16E-10	1.23E-10	1.23E-10
TOTAL								**4.37E-10**	**9.65E-10**	**8.74E-11**	**1.93E-10**

Table 3.4-1 (Cont'd)
Risk Impact Analysis Results

System[1]	Safety Significance	Break Location[2]	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank[3]	SXI[4]	RIS_B	Delta	w/ POD	w/o POD	w/ POD	w/o POD
MFW	High	MFW – 1	None (FAC)	Low (High)	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MFW	High	MFW – 1	None	Low	10	6	-4	2.00E-12	2.00E-12	4.00E-13	4.00E-13
MFW	High	MFW – 2	None	Low	0	2	2	-1.65E-11	-1.65E-11	-3.30E-12	-3.30E-12
MFW	High	MFW – 3	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MFW	High	MFW – 4	None	Low	2	2	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL								**-1.45E-11**	**-1.45E-11**	**-2.90E-12**	**-2.90E-12**
MS	High	MS – 2	TT	Medium	2	1	-1	-6.00E-12	1.00E-11	-1.20E-12	2.00E-12
MS	High	MS – 2	None	Low	2	0	-2	1.00E-12	1.00E-12	2.00E-13	2.00E-13
MS	High	MS – 3	None	Low	4	0	-4	2.00E-12	2.00E-12	4.00E-13	4.00E-13
MS	Low	Class 2 LSS	N/A	Assume Medium	18	0	-18	5.04E-10	5.04E-10	1.01E-10	1.01E-10
TOTAL								**5.01E-10**	**5.17E-10**	**1.00E-10**	**1.03E-10**
EFW	Low	Class 2 LSS	N/A	Assume Medium	1	0	-1	2.80E-11	2.80E-11	5.60E-12	5.60E-12
TOTAL								**2.80E-11**	**2.80E-11**	**5.60E-12**	**5.60E-12**
RBS	Low	Class 2 LSS	N/A	Assume Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL								**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
GRAND TOTAL								**-3.16E-09**	**2.26E-09**	**-6.33E-10**	**4.53E-10**

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.

Table 5
Inspection Location Selection Comparison Between ASME Section XI Code and Code Case N-716

System(1)	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N-716	
	High	Low		DMs	Rank(2)			Vol/Sur	Sur Only	RIS_B	Other(3)
RC	✓		LOCA	TASCS, TT, (PWSCC)	Medium (Medium)	B-F	1	1	0	0	–
						B-J^{DMW}	2	0	0	2	–
RC	✓		LOCA	TASCS, TT	Medium	B-J	25	5	6	7	–
RC	✓		LOCA	TT, (PWSCC)	Medium (Medium)	B-F	3	3	0	1	–
RC	✓		LOCA	TT	Medium	B-J	3	0	1	1	–
RC	✓		LOCA	None (PWSCC)	Low (Medium)	B-F	3	3	0	1	–
RC	✓		LOCA	None	Low	B-J^{DMW}	11	0	0	0	–
						B-J	117	20	2	7	–
RC	✓		PLOCA	None	Low	B-J	23	0	3	0	–
MUP	✓		PLOCA	TT, IGSCC	Medium	B-J	4	0	0	1	–
MUP	✓		LOCA	TT	Medium	B-J^{DMW}	4	3	1	1	–
						B-J	87	8	29	15	–
MUP	✓		PLOCA	IGSCC	Medium	B-J	5	0	0	1	–
MUP	✓		LOCA	None	Low	B-J^{DMW}	1	0	0	0	–
						B-J	7	0	3	0	–
MUP	✓		PLOCA	None	Low	B-J	67	0	0	0	–
MUP		✓	Class 2 LSS	N/A	Assume Medium	C-F-1	705	65	0	0	–
DH	✓		LOCA	TT	Medium	B-J	7	4	0	2	–
DH	✓		LOCA	None (PWSCC)	Low (Medium)	B-J^{DMW}	1	0	0	1	–
DH	✓		LOCA	None	Low	B-J	21	6	0	4	–
DH	✓		PLOCA – SD	None	Low	B-J	25	4	2	0	–
DH	✓		SD	None	Low	C-F-1	90	25	0	8	–
DH		✓	Class 2 LSS	N/A	Assume Medium	C-F-1	451	22	0	0	–

Table 5 (Cont'd)
Inspection Location Selection Comparison Between ASME Section XI Code and Code Case N-716

System[1]	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N-716	
	High	Low		DMs	Rank[2]			Vol/Sur	Sur Only	RIS_B	Other[3]
MFW	✓		MFW – 1	None (FAC)	Low (High)	C-F-2	6	0	0	0	–
MFW	✓		MFW – 1	None	Low	C-F-2	81	10	0	6	–
MFW	✓		MFW – 2	None	Low	C-F-2	4	0	0	2	–
MFW	✓		MFW – 3	None	Low	C-F-2	6	0	0	0	–
MFW	✓		MFW – 4	None	Low	NNS	2	2	0	2	–
MS	✓		MS – 2	TT	Medium	C-F-2	2	2	0	1	–
MS	✓		MS – 2	None	Low	C-F-2	2	2	0	0	–
MS	✓		MS – 3	None	Low	NNS	4	4	0	0	–
MS		✓	Class 2 LSS	N/A	Assume Medium	C-F-2	130	18	0	0	–
EFW		✓	Class 2 LSS	N/A	Assume Medium	C-F-2	8	1	0	0	–
RBS		✓	Class 2 LSS	N/A	Assume Medium	C-F-1	176	0	0	0	–

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-1 RIS_B application. The "Other" column has been retained in this table solely for uniformity purposes with other RIS_B application template submittals.

Attachment 2 to

1CAN060902

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, if warranted.	✓		June 30, 2011
Request for Alternative CEP-ISI-007 pertaining to the application of Code Case N-663 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
Consistent with previously approved RI ISI submittals, ANO-1 will calculate coverage and use additional examinations or techniques in the same manner it has 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
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
Final Materials Reliability Program (MRP) guidance on the subject of TASCS will be incorporated into the ANO-1 RIS_B application, if warranted.	✓		Upon NRC approval of this request for alternative