VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

July 25, 2001

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

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Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
RISK-INFORMED INSERVICE INSPECTION PROGRAM

In a letter dated April 26, 2001, Virginia Electric and Power Company (Dominion) submitted a RI-ISI Program for North Anna Units 1 and 2 and the associated relief request for review and approval as an alternative to current ASME Section XI inspection requirements for Class 1 piping. In a July 9, 2001 telephone conference call, the NRC staff requested additional information to complete the review of the proposed RI-ISI program. The attachment to this letter provides the requested information.

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

Very truly yours,

Leslie N. Hartz
Vice President - Nuclear Engineering and Services

Commitments made in this letter:

1. None

Attachment

A047

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Response to Request for Additional Information for the Risk-Informed Inservice Inspection (RI-ISI) Program for ASME Class 1 Piping

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 and 2

REQUEST FOR ADDITIONAL INFORMATION ON THE NORTH ANNA UNITS 1 AND 2 RISK-INFORMED INSERVICE INSPECTION PROGRAM PLAN

Question 1

One major step in the WCAP process is the identification of degradation mechanisms and the development of corresponding pipe failure frequencies. The requested Table 1 summarizes the qualitative results of this step by identifying the different degradation mechanisms, combinations of mechanisms, and the prevalence of the different mechanism. The calculated ranges in Table 1 summarize the quantitative results of the analysis. This information will illustrate how the degradation mechanism identification and failure frequency development step in the WCAP methodology was implemented, and provide an overview of the results generated. Please expand the current Table 3.4-1 to include the following information.

a) System	b) Degradation Mechanism/ Combination	c) Failure Probability Range at 40 years with no ISI		d) Number of Susceptible Segments	e) Comments
		Leak	Disabling Leak		

a) System: Each system included in the analysis.

b) Degradation Mechanism/Combination: Segment failure probabilities are characterized in the WCAP method by imposing all degradation mechanisms in a segment (even if they occur at different welds) and the worst case operating conditions at the segment on a “representative” weld, and using the resulting failure probability for the segment. Please identify the dominant degradation mechanisms and combination of degradation mechanisms selected in each system. The reported mechanisms should cover all segments in the system. The table in the current submittal is not clear about which specific degradation mechanisms or combination of mechanisms are included in the leak estimates provided.

c) Failure Probability Range at 40 years with no ISI: For each dominant degradation mechanism and combination of degradation mechanisms, please provide the range of estimates developed for the leak and disabling leak sizes as applicable. The table in the current template provided the range of leak estimates only.

d) Number of Susceptible Segments: Please identify the total number of segments susceptible to each dominant degradation mechanism and combination of degradation mechanisms.

e) Comments: The contents of this column are still being developed. It should provide further explanation and clarifications on the degradation mechanism and results as appropriate. Examples of items to be included are identification of which degradation mechanism are applied to socket welds and if a break calculation was needed to evaluate pipe whip constraints.

Response:
Each unit was evaluated for failure probability and these results are detailed below. No break calculations were necessary for this Class 1 evaluation on any system. However, they were calculated and are available.

a) System	*b) Degradation Mechanism/ Combination*	*c) Failure Probability Range at 40 years with no ISI*		*d) Number of Susceptible Segments*	*e) Comments*
		Leak	*Disabling Leak*		
ACC	*fatigue(default, e.g., no mechanism, snubber locking up in thermal conditions)*	*U1 3.46E-5 to 6.30E-5* *U2 3.44E-5 to 4.40E-4*	*U1 1.55E-5 to 5.37E-5* *U2 2.20E-5 to 3.70E-4*	*U1 - 9* *U2 - 9*	*Includes both butt and socket welds*
CH	*vibratory fatigue*	*U1 3.86E-3 to 6.81E-3* *U2 3.87E-3 to 6.81E-3*	*U1 2.60E-3 to 1.24E-2* *U2 2.61E-3 to 1.24E-2*	*U1 - 9* *U2 - 9*	*Includes both butt and socket welds, vibratory fatigue attributed to segments based upon proximity to RC Pump*
CH	*fatigue(default, e.g., no mechanism, snubber locking up in thermal conditions)*	*U1 3.65E-7 to 1.50E-3* *U2 2.88E-6 to 7.52E-4*	*U1 2.43E-7 to 5.62E-4* *U2 2.15E-6 to 5.61E-4*	*U1 - 34* *U2 - 34*	*Includes both butt and socket welds*
ECC	*fatigue(default, e.g., no mechanism, snubber locking up in thermal conditions)*	*U1 8.66E-7 to 7.52E-4* *U2 8.66E-7 to 7.52E-4*	*U1 3.84E-7 to 3.98E-4* *U2 3.82E-7 to 5.15E-4*	*U1 - 50* *U2 - 49*	*Includes both butt and socket welds*

RC	*vibratory fatigue*	*U1 2.50E-5 to 6.83E-3* *U2 2.50E-5 to 6.75E-3*	*U1 2.58E-5 to 4.46E-3* *U2 2.58E-5 to 4.48E-3*	*U1 - 20* *U2 - 20*	*Includes both butt and socket welds, vibratory fatigue attributed to segments based upon proximity to RC Pump*
RC	*fatigue(default, e.g., no mechanism, snubber locking up in thermal conditions)*	*U1 9.82E-8 to 1.59E-4* *U2 9.57E-8 to 1.59E-4*	*U1 3.80E-8 to 1.10E-4* *U2 3.67E-8 to 4.42E-5*	*U1 - 93* *U2 - 91*	*Includes both butt and socket welds, augmented program (HELB) affects two segments on Unit 1 and three segments on Unit 2*
RC	*Striping /Stratification*	*U1 6.92E-5 to 9.07E-5* *U2 7.35E-5 to 9.07E-5*	*U1 2.12E-5 to 4.06E-5* *U2 2.18E-5 to 4.06E-5*	*U1 - 6* *U2 - 6*	*Limited to butt welds, these segments are associated with safety injection to the hot legs and cold legs*
RH	*fatigue(default, e.g., no mechanism, snubber locking up in thermal conditions)*	*U1 4.44E-6 to 7.79E-6* *U2 4.44E-6 to 9.16E-5*	*U1 3.46E-6 to 6.63E-6* *U2 3.62E-6 to 3.28E-5*	*U1 - 3* *U2 - 7*	*Includes both butt and socket welds*

Note: ACC (SI) – Accumulator, CH – Chemical & Volume Control, ECC (SI) – Emergency Core Cooling, RC – Reactor Coolant, and RH – Residual Heat Removal

Question 2

Another major step in the WCAP process is assignment of segments into safety significance categories based on an integrated decision making process, and the selection of segments for inspection locations. The requested Table 3 summarizes the results of the safety significance categorization process as determined by the quantitative criteria, by the expert panel's deliberation on the medium safety significant segments, and by the expert panel's deliberations based on other considerations. The summarizing information requested in Table 3 will provide an overview of the distribution of the safety significance of the segments based on the quantitative results, and the final distribution based on the integrated decision making. Each segment has four RRWs calculated, a CDF with and without operator action, and a LERF with and without operator action. Please provide the following Table.

System	Number of Segments with Any RRW >1.005	Number of Segments with Any RRW Between 1.005 and 1.001	Number of Segments with Any RRW Between 1.005 and 1.001 Placed in HSS	Number of Segments with All RRW < 1.001 Selected for Inspection

Response:
The table below reflects the information at the time of the Expert Panel Meeting based upon the minutes of the meeting. In a few instances the Expert Panel disagreed with the quantitative results presented and requested new quantification based on different assumptions. As a result the final RRW calculations may vary from the results presented to the Expert Panel. The differences were determined to be minor or supportive of the Expert Panel determinations. Additionally, the Expert Panel determined conservatively that segments with exact value RRW of 1.005 were quantitatively HSS and the table below has been modified accordingly.

System	*Number of Segments with Any RRW ≥ 1.005*	*Number of Segments with Any RRW Between 1.005 and 1.001*	*Number of Segments with Any RRW Between 1.005 and 1.001 Placed in HSS*	*Number of Segments with All RRW < 1.001 Selected for Inspection*
ACC	*0*	*0*	*0*	*0*
CH	*6*	*0*	*0*	*6*
ECC	*16*[1]	*3*	*2*	*3*
RC	*38*	*16*	*7*	*9*
RH	*3*[2]	*0*	*0*	*0*

Note 1: Segment ECC-037 was originally presented as high but was corrected for modeling error by Expert Panel and made LSS.
Note 2: Segments RH-004 and RH-007 exist on Unit 2 only.
Note: ACC (SI) – Accumulator, CH – Chemical & Volume Control, ECC (SI) – Emergency Core Cooling, RC – Reactor Coolant, and RH – Residual Heat Removal

Question 3
Another major step in the WCAP process is development of the consequences of segment ruptures. The WCAP methodology requires that a summary of the consequences be developed for each system and provided to the expert panel during their deliberations. Please provide this summary for each system. The summary will illustrate that the appropriate types of consequences (i.e., initiating events, mitigating

system failure, and combinations) are included in the evaluation and will provide an overview of the results of the step.

Response:
As presented to the Expert Panel:

ACC (SI) – The direct consequences modeled were loss of one accumulator and its' applicable loss of one RHR flow path. There were no indirect consequences modeled.

ECC (SI) – The direct consequences modeled were loss of RWST, loss of cold leg injection from the high head pumps, and loss of cold and hot leg injection from the low head pumps as applicable to the segment. The PSA model does not model high head hot leg injection but does include hot leg recirculation. Additionally, the individual loop flow paths are not modeled. It was assumed that loss of one loop of SI injection was insignificant due to redundancy. No indirect effects were modeled.

CH – The direct consequences postulated for the CH system were primarily: Loss of charging, loss of seal injection, loss of emergency boration function. No indirect consequences were assumed.

RC – The direct consequences modeled were associated with LOCAs (large, medium, and small) as an initiating event. The model assumed large pipe could have all three type LOCAs, medium pipe both medium and small LOCAs and small pipe only small LOCAs. No indirect consequences were assumed. In general consequence is high.

RH – The direct consequences modeled were associated with the loss of the RH system. This has a significant impact in the PSA model with regard to steam generator tube rupture. No indirect consequences were assumed.

Question 4
Please add the statement that the sensitivity study to address uncertainty as described on page 125 (Section 3.6.1) of WCAP-14572, Rev. 1-NP-A was performed and identify how many segments' RRW increased from below 1.001 to greater than or equal to 1.005. If the sensitivity study was not performed, provide a description and justification of any deviation.

Response:
The uncertainty analysis as described on WCAP page 125 was performed and is now included as part of the base process of the risk evaluation. As a result of the uncertainty analysis, no segments' RRW increased from below 1.001 to greater than or equal to 1.005.

Question 5
Please state that the change in risk calculations were performed according to all the guidelines provided on page 213 (Section 4.4.2) of WCAP-14572, Rev. 1-NP-A or provide a description and justification of any deviation.

Response:
The change in risk calculation was performed according to the guidelines provided on page 213 of the WCAP with one deviation. The proposed program is Class 1 only, which includes the RCS or systems directly connected to the RCS. These systems are monitored for leakage extensively, consequently credit for leak detection was applied to all the segments in the North Anna program for the change in risk analysis both inside and outside containment. The calculation should have applied the leak detection to segments inside containment only. Subsequently, the calculation was performed as a sensitivity study with leak detection excluded from segments outside containment. No change in selections resulted because the resultant values varied only slightly.

Question 6
The quantitative change in risk results are adequately summarized in the current template tables 3-5 and 3-10. Please state that all four criteria for accepting the final selection of inspection locations provided on pages 214 and 215 (Section 4.4.2) of WCAP-14572 Rev. 1-NP-A were applied. If all four criteria were not used, please provide a description and justification of the deviation. If comparison with any of the criteria indicated that "reevaluation" of the selected locations was needed, please identify the criteria that required the reevaluation and summarize the results of the reevaluation. If the results of the reevaluation can be found in the footnotes of Table 5-1, please refer to the footnotes.

Response:
All four criteria for accepting the results discussed on page 214 and 215 in the WCAP were applied. No reevaluation was required.

Question 7
Briefly describe the qualifications, experience, and training of the users of the SRRA code on the capabilities and limitations of the code.

Response:
An engineering team was established that had access to expertise from ISI, NDE, materials, stress analysis and system engineering. The team was trained in the failure probability assessment methodology and the Westinghouse structural reliability and risk assessment (SRRA) code, including identification of the capabilities and limitations as described in WCAP-14572, Revision 1-NP-A, Supplement 1. Also the team members had participated on the engineering team for the Surry Unit 2 RI-ISI project or both the Surry Unit 1 and Unit 2 RI-ISI projects previously approved by the NRC Staff.

Question 8

Please provide the following information regarding the treatment of augmented programs during the RI-ISI program development.

a) Treatment of augmented program inspections during categorization is described on page 80 (Section 3.5.5) of WCAP-14572, Rev. 1-NP-A. Page 5 of your submittal states that, "another consideration was whether a segment is included in the plant high-energy line break (HELB) augmented program." Please explain how this information was used to determine which failure probability was used in the risk-informed ISI program and how this comports with the WCAP Topical Methodology.

Response:
The North Anna RI-ISI proposed program addressed the existing augmented program on the Class 1 systems. The UFSAR required augmented inspection program was not affected by the proposed RI-ISI program. The effects of ISI from the existing augmented program are included in the risk evaluation used to assist in categorizing the segments as described on page 80 (section 3.55 of the WCAP). The HELB program required by the UFSAR provides volumetric examination of certain weld locations within the RC system. WCAP-14572, Rev. 1-NP-A requires on page 105 in section 3.6.1 that for piping segments that are included in augmented programs (such as erosion-corrosion and stress corrosion cracking programs), the SRRA failure probabilities with ISI but without leak detection be used. This approach is again applied in the change in risk calculation (ref. WCAP section 4.4.2, page 213) for segments which have an augmented program with the exception that credit for leak detection is now applied. Both of these requirements of the WCAP were followed for North Anna.

b) When the SRRA code is used for calculating failure probabilities for FAC, please describe if calculations were coordinated with the existing plant program since the code requires input that can be obtained from the knowledge gained from ongoing monitoring and evaluations of wall thinning rates.

Response:
North Anna's RI-ISI proposed programs are limited to Class 1 systems. The FAC damage mechanism is not postulated, since the systems involved do not operate under conditions conducive to FAC. Additionally, the systems are not fabricated from materials susceptible to FAC.

Question 9

Please confirm that SRRA code was only used to calculate failure probabilities for the failure modes, materials, degradation mechanisms, input variables, and uncertainties it was programmed to consider as discussed in the WCAP Supplement 1, page 15. For example, the SRRA code should only be applied to standard piping geometry (circular piping geometry with uniform wall thickness). If the code was applied to any non-standard geometry, please describe how the SRRA inputs were developed.

Response:
The SRRA code was used to calculate failure probabilities for the failure modes, materials, degradation mechanisms, input variables and uncertainties it was programmed to consider as discussed in the WCAP Supplement 1. All the piping configurations included in the RI-ISI program could be adequately modeled using the SRRA code.

Question 10
Please describe any sensitivity studies performed to support the use of the SRRA code.

Response:
The engineering team assesses industry and plant experience, plant layout, materials, and operating conditions and identifies the potential failure mechanisms and causes. Information was gathered from various sources by the Engineering team to provide input for the SRRA model. Sensitivity studies were performed to aid in determining representative input values when sufficient information was not available. Snubber failure history was also reviewed to identify any potential effects that could increase piping failure probability. The resulting failure probabilities were compared against the postulated damage mechanisms and industry/plant experience for reasonableness. Examples include the expectation of higher failure probabilities for vibratory fatigue and thermal fatigue from striping or stratification, and lower failure probabilities for no active mechanism (default thermal fatigue). These failure probabilities were affected by pipe size, assumed stresses and initial construction inspection requirements. Consequently, within each type of postulated damage mechanism smaller pipe tended to have higher failure probabilities, as well as piping that did not receive volumetric examination at the time of construction (socket welds). These type results were expected by the team.

Question 11 Intentionally left blank.

Question 12
Please summarize the system design features and other physical characteristics of the plant as reflected in the risk evaluations that determined the location and the number of locations selected for inspection.

Response:
LOCA initiation was the primary consequence of failure within the Class 1 boundary. This was numerically evident in the risk evaluation results and a primary focus of the Expert Panel. The selections for the RI-ISI program are geared to support prevention of this type of event. Additionally, loss of safety injection water outside of containment was a concern and inspection was designated for these areas. Finally, Alloy-600 type welds were identified on the Steam Generator and Pressurizer piping connections. These areas were also selected for examination.

Question 13
Section 3.4 of your submittal states that, "The engineering team that performed this evaluation used the Westinghouse structural reliability and risk assessment (SRRA)

software program ... to aid in the process." Page 83 (Section 3.5.6) of WCAP-14572, Rev. 1-NP-A states that for WOG plant application "(SRRA) tools were used to estimate the failure probabilities for the piping segment". Pages 6 and 7 of the related safety evaluation (SE Section 3.2.3) also state that the failure probability estimate of the weld "is subsequently used to represent the failure probability of the segment." Please explain how the quantitative SRRA results were used and how your method comports with WCAP-14572, Rev. 1-NP-A and the associated SE. If the quantitative results were not directly used as input into the calculations, please describe the experience and training of the team members that selected the values for use in the calculations.

Response:
The failure probabilities for the North Anna piping segments were all derived using the Westinghouse Windows version of the structural reliability and risk assessment (SRRA) software program. As such, no deviation from the methodology described in WCAP-14572, Rev. 1-NP-A and the associated SE were made.

Question 14
Page 7 of your submittal reports that there are 26 segments in Region 1 and 64 segments in Region 2. The submittal further states that one segment in Region 1 and 46 segments in Region 2 were evaluated using the Perdue Model.

a) Why was the statistical method not applicable to 25 Region 1 and 18 Region 2 segments?

   Response:
   The segments identified required a visual, VT-2 examination. There were 19 Region 1 segments postulated with vibratory fatigue, and 6 Region 1 segments comprised of socket welds under relief request R-1. Also there were 6 Region 2 segments postulated with vibratory fatigue, and 12 Region 2 segments comprised of socket welds under relief request R-1. The visual, VT-2 examination inspects the entire segment for leakage at pressure. Therefore, a minimum number of specific examination locations is not required (ref. WCAP-14572, Rev. 1-NP-A, section 3.7.3, page 184). The socket-welded segments typically terminate at a branch connection, which is included in the segment. Consequently, the branch connection receives the same VT-2 examination.

b) How was the number of elements to be inspected determined for the 25 Region 1 and 18 Region 2 segments not evaluated by the statistical method?

   Response:
   Since the examination required is a visual, VT-2 examination the entire segment is examined in each case.

c) What size of piping and type of welds are the 25 segments in Region 1 not evaluated by the statistical method? What were the accident sequences used to represent the consequence of rupture for these segments?

Response:

Segment #	*Postulated Damage Mechanism*	*Size*	*Weld Type*	*Accident Sequence Postulated*
CH-001	*Vibratory Fatigue*	*1 ¼" to 2"*	*Socket and Butt*	*IE - Small LOCA*
CH-002	*Vibratory Fatigue*	*1 ¼" to 2"*	*Socket and Butt*	*IE - Small LOCA*
CH-003	*Vibratory Fatigue*	*1 ¼" to 2"*	*Socket and Butt*	*IE - Small LOCA*
CH-012	*Vibratory Fatigue*	*¾"*	*Socket, branch connection*	*IE - Small LOCA*
CH-013	*Vibratory Fatigue*	*¾"*	*Socket, branch connection*	*IE - Small LOCA*
CH-014	*Vibratory Fatigue*	*¾"*	*Socket, branch connection*	*IE - Small LOCA*
ECC-036	*Default Thermal Fatigue (Snubber Locking Up)*	*¾"*	*Socket , branch connection*	*SYS - Loss of Unit RWST Inside Containment, and Loss of Alternate Path of Hi-Head SI to Cold Leg*
ECC-041	*Default Thermal Fatigue (Snubber Locking Up)*	*¾" to 1"*	*Socket, branch connection*	*SYS – Loss of RWST Outside Containment*
ECC-043	*Default Thermal Fatigue (Snubber Locking Up)*	*¾"*	*Socket, branch connection*	*SYS – Loss of RWST Outside Containment and Loss of Low Head SI to Cold Legs*
ECC-044	*Default Thermal Fatigue (Snubber Locking Up)*	*¾"*	*Socket , branch connection*	*SYS – Loss of Low Head SI to Hot Leg From Low Head SI Train A*
ECC-045	*Default Thermal Fatigue (Snubber Locking Up)*	*¾"*	*Socket, branch connection*	*SYS – Loss of Low Head SI to Hot Leg From Low Head SI Train B*
ECC-048	*Default Thermal Fatigue (Snubber Locking Up)*	*¾"*	*Socket, branch connection*	*SYS - Loss of Unit RWST Inside Containment, and Loss of Alternate Path of Hi-Head SI to Cold Leg*

Segment #	Postulated Damage Mechanism	Size	Weld Type	Accident Sequence Postulated
RC-031	Vibratory Fatigue	2"	Socket, Butt, and branch connection	IE - Small LOCA
RC-032	Vibratory Fatigue	2" to 3"	Socket, Butt, and branch connection	IE - Small LOCA
RC-033	Vibratory Fatigue	2" to 3"	Socket, Butt, and branch connection	IE - Small LOCA
RC-037	Vibratory Fatigue	2"	Socket, branch connection	IE - Small LOCA
RC-038	Vibratory Fatigue	2"	Socket, branch connection	IE - Small LOCA
RC-039	Vibratory Fatigue	2"	Socket, branch connection	IE - Small LOCA
RC-066	Vibratory Fatigue	¾"	Socket, branch connection	IE - Small LOCA
RC-067	Vibratory Fatigue	¾"	Socket, branch connection	IE - Small LOCA
RC-068	Vibratory Fatigue	¾"	Socket, branch connection	IE - Small LOCA
RC-069	Vibratory Fatigue	¾"	Socket, branch connection	IE - Small LOCA
RC-075	Vibratory Fatigue	¾"	Socket, branch connection	IE - Small LOCA
RC-076	Vibratory Fatigue	¾"	Socket, branch connection	IE - Small LOCA
RC-077	Vibratory Fatigue	¾"	Socket, branch connection	IE - Small LOCA

Note: IE – Initiating Event Treatment, SYS – System Event Treatment, LOCA – Loss of Coolant Accident, RWST – Refueling Water Storage Tank, SI – Safety Injection

Question 15
Page 5 of your submittal stated that information on whether a segment is included in the high-energy line break augmented program was used to determine which failure probability was used in the risk-informed ISI process. Please explain how the HELB program was used to modify the segment failure probabilities.

See response to question 8a.

Question 16
The staff safety review of the December 14, 1992, North Anna IPE submittal noted that human errors related to calibration of equipment were not well treated in the original HRA. Please identify how calibration errors are treated in the PRA used to support the RI-ISI submittal and provide a justification that this treatment is adequate to support the risk ranking and the change in risk conclusions. Do the results of the WOG peer review

for the Surry plant (discussed in your North Anna RI-ISI submittal) address this issue? If so, what observations would be valid for the North Anna PRA?

Response:
Calibration errors are treated in the PRA model used to prepare the RI-ISI submittal. The North Anna PRA model contains instrument channel common cause failure basic events for eight key RPS and ESFAS inputs. This modeling of the common cause failure implicitly includes human error related to calibration of equipment. As can be seen from the table below the common cause failure of these channels is not risk significant in the base PRA model.

Basic Event	*Description*	*RAW*	*RRW*
1FWLIC-CC-SGLEV	*2/3 SG Narrow Range Level Instrument Channels*	*1.04*	*1.00*
1LMPIC-CC-100	*3/4 Containment Pressure Channels*	-	-
1MSFIC-CC-MSFLOW	*2/3 Main Steam Flow Instrument Channels*	-	-
1MSPIC-CC-MSLP	*2/3 Main Steam Low Pressure Instrument Channels*	-	-
1MSPIC-CC-STMDPR	*Steam Differential Pressure Instrument Channels*	-	-
1RCPIC-CC-PRSZRP	*Pressurizer Pressure Instrument Channels*	-	-
1RCTIC-CC-TAVG	*Tavg Temperature Instrument Channels*	-	-
1SILIC-CC-RWST	*RWST Level Instrument Channels*	*1.92*	*1.00*

This treatment is adequate to support the pipe segment risk ranking for the following reason. The pipe segments are risk ranked both with and without operator error. What that means is segment risk is calculated twice. In the first calculation we take no credit for operator actions and in the second calculation we assume the operator actions guarantee success. If a segment is risk significant from either perspective, then it is ranked high. This approach bounds the question about the numerical value of a Human Error Probability (HEP) since both ends of the spectrum are considered.