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U.S. Nuclear Regulatory Commission
Document Control Desk
Office of Nuclear Reactor Regulation
Washington, D C. 20555-0001

Ref (1) Letter, Robert G Schaaf (NRC) to Robert C. Mecredy (RG&E), December 26, 2002 "Request for Additional Information Regarding Severe Accident Mitigation Alternatives for the R. E Ginna Nuclear Power Plant"

Subject Clarifications to 1/31/03 and 2/28/03 SAMA RAI Responses
R. E Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr Schaaf

This letter is in response to a telecon held on March 28, 2003 regarding some clarifications requested relative to RG&E's January 31 and February 28, 2003 responses to Reference (1) These clarifications are provided in the attachments. Any questions relating to these responses should be directed to our License Renewal Project manager, George Wrobel, at (585) 771-3535.

Very truly yours,

Robert C. Mecredy

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Attachments

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Attachment 1

Clarification 1:

The process to identify potential SAMAs was not clearly articulated in the Environmental Report RAI Question 1 attempted to extract some details. In the January 31st response to RAI 1a, RG&E stated the Ginna focuses the process of identifying SAMAs by concentrating on those events that are dominant contributors to CDF and LERF, and provided data such as importance analysis results In the response to RAI 1d, RG&E states that it reviewed many SAMAs evaluated in recent license renewal applications to determine their potential applicability to Ginna. In the revised SAMA evaluation (Attachment 2 to the February 28, 2003 response), it states that RG&E reviewed SAMA analyses submitted in support of license renewal activities for other nuclear power plants and NRC and industry documentation discussing potential plant improvements; however, which plants and which industry documents were used were not identified.

Please indicate which plants' SAMA evaluations and which NRC and industry documents were used to identify potential SAMAs. Also, indicate an approximation of how many SAMAs were reviewed

Response to Clarification 1:

The process RG&E used to identify SAMAs was discussed in the responses to RAI 1 included in both the January and February RAI response submittals. The following discussion provides further clarification for the criteria used in the identification process.

RG&E supplemented their initial efforts to identify potential SAMAs by considering modifications evaluated by previous applicants, in particular those considered by the Omaha Public Power District (OPPD) for Fort Calhoun Station (FCS). These additional modifications that were considered (192 total) include those considered by previous applicants (submittals dated prior to January 2002), as well as modifications specific to FCS.

RG&E's review of these modifications, as well as the identification of site-specific concepts, was focused by the following criteria:

1. Applicability to Ginna in general

2. Applicability to Ginna Station-specific areas of "high" risk significance

3. Potential for significant risk reduction

4. Excessive cost, defined as greater than 2 times the maximum attainable benefit (MAB) The MAB is a very conservative value that represents the theoretical monetized benefit

of eliminating all plant risk. RG&E acknowledges that significant margin exists in this value, illustrated by the fact that no single modification can eliminate all plant risk. Initial ideas (~20) were then further defined and reviewed in terms of the following criteria:

1. Ability to implement change (i.e., design challenges/physical limitations)

2. Whether implementation would increase vulnerabilities in other areas

3. Realistic risk reduction potential

This process resulted in the eight plant-specific candidate modifications evaluated in the SAMA analysis. The increase in MAB ($992,000 to $1,928,000), as a result of revising the analysis consistent with the Ginna Station PSA Rev. 4.2, did not result in the identification of additional SAMA candidates for evaluation.

Attachment 2

Clarification 2:

In response to RAI 2b (January 31, 2003 letter), RG&E states that all vulnerabilities and items of concern were resolved, from the IPEEE, except for seismically induced flooding resulting from the failure of the Reactor Makeup Water Tank and the Monitor Tank. RG&E states that since it is evaluating potential modifications to address this issue, it was not addressed further in the SAMA analysis In response to RAI 4d, RG&E states that a variety of hardware modifications are being evaluated to resolve the issue Please discuss the potential modifications under consideration, estimated costs, and benefits.

Response to Clarification 2:

One conceptual modification to resolve this IPEEE open item is to increase the height of the curb around the RHR sub-basement entrance to a level that would neither pose a flooding danger to the SI pumps, nor allow the RMWT and monitor tank contents to enter the sub-basement. Another conceptual design is to provide a dam around the RMWT to channel any water escaping the tank to outside the Auxiliary Building. Other concepts are also being explored. These are being tracked by our Commitment and Action Tracking System (CATS item 10602).

Attachment 3

Clarification 3:

In NUREG-1742 for the Ginna plant, it states that 52 items of equipment could not be screened out, and some 90 items of equipment were identified as being vulnerable to block walls. In response to RAI 4b (January 31, 2003 letter), RG&E refers to the 52 outliers, but does not address the 90 pieces of equipment that are vulnerable to block walls. Please indicate what the status is of these 90 vulnerable pieces of equipment. Also, according to the response, RG&E internally evaluated the 52 outliers and through a combination of reanalysis, procedure changes, and modifications, addressed the outliers. Please discuss details of the resolution of these 52 outliers

Response to Clarification 3:

NUREG-1742 is not current since several of these noted vulnerabilities have previously been addressed. As stated in the NRC's February 25, 1999 "Request for Additional Information on the R.E. Ginna Nuclear Power Plant IPEEE submittal", Section A. Seismic, "...no further actions are planned for 38 of the 52 outliers that are not related to A-46 closeout " However, RG&E did resolve all of these outliers. The resolution of those 38 items is provided below.

Also, the status of the 90 pieces of equipment vulnerable to block walls was provided in Enclosure 1, Attachment C of our July 30, 1999 IPEEE submittal.

Status of 38 IPEEE Outliers

1.	Valve 1789	PCR 98-052 Rev. 1 installed valve operator restraint
2,3	Valves 4297, 4298	DA-ME-99-014 analyzed stresses to be acceptable
4-6	Valves 4770 C,D,E	PCR 98-048 installed anchorage modification
7-14	Dampers 5871-5877-5880	PCR 2000-011 installed two new duct hangers
15,16	Valves 830A, 830B	PCR 98-091 modified supports
17-20	Valves 875A/B, 876A/B	SEV 1087 analyzed function of these valves as not safety-related
21-24	Air handling units ACP01-04	PCR 98-065 modified anchorage
25,26	Charcoal filter units ACP06, ACP07	PCR 98-065 modified anchorage
27,28	CCW HX EAC01A/B	PCR 97-059 upgraded support saddles
29,30	RHR HX EAC02A/B	DA-ME-97-057 analyzed supports as acceptable
31-33	Instrument rack doors	PCR 98-044 installed locks
34,35	RHR Pumps PAC01A/B	PCR 99-061 modified anchorage
36-38	Air handling units AKF07, AKF08, AKP02	PCR 99-077 modified anchorage

Attachment 4

Clarification 4

In response to RAI 3 (February 28, 2003 letter), RG&E provides tabular lists of Source Term Release Bins, Fractional Fission Product Releases, and Source Term Release Category Bins. However, it is not clear from the information provided how some of the IPE Source Term Categories are related to the bins For example, IPE STCs 2, 3, 4, 5, 6, 8, 9, 11, 12, 14, 15, 17, 18, 20, and 21 are not listed in Tables 2 and 4. Please provide a statement relative to the contribution (or not) to the 11 identified bins. If a bounding STC was used to represent several STCs, indicate which STCs were grouped together.

Response to Clarification 4

In the Ginna source term categorization from the original IPE submittal (currently found in Appendix I of Reference 1) all major release categories were mapped to one of 25 source term classes (STCs) with a representative release profile. As the simplified Level 2 PSA analysis, as currently used, bins events into 11 classes, only 11 of the available 25 source term classes were used. This condensation was performed so as to conserve the class release frequency and to provide a single representative release associated with that class. With the exception of intact containment state 21 which is identical to state 1 (See Table I-18 of Reference 1), the remaining 13 source term classes are subsumed into 5 of the final 11 classes, the sum of whose frequencies comprise about 7% of the plant CDF. The distribution of the STCs is summarized in Table 1

1 R. E. Ginna Power Plant Probabilistic Safety Analysis, Final Report, Revision 4.0, February, 2002

Table 1: Mapping of Source Term Release Classes				
Release Category Bin**	Contribution to CDF yr^{-1}	%CDF*	Represented by STC #	Other STC's in class
Intact	2.38E-05	59.80	1	21
SGTR-Wet	4.53E-06	11.60	24	None
SGTR- ARV Cycle	1.32E-06	3.32	23	None
LATE_Small	1.05E-06	2.64	13	11, 12, and 17
LATE_Global	1.05E-06	2.64	10	14
LOCI	4.17E-07	1.05	19	18
HPRCS	4.13E-07	1.03	16	2, 3, 4,5, 6, 15
ISLOCA	2.5E-07	0.63	22	None
TISGTR	2.48E-08	0.06	25	None
LPRCS	7.95E-09	0.02	7	8, 9, 20

* Percentage based on total CDF that includes shutdown contribution. This contribution is not shown in this table and represents about 17% of the total CDF.

**Only ten of the eleven classes are explicitly presented. The frequency contribution to one bin was negligible and it has been omitted from the table.

Attachment 5

Clarification 5:

In response to RAI 7I, RG&E provided a brief discussion of the RCP Seal LOCA. It is unclear from this discussion and from the discussion contained in the IPE as to the RCP seal LOCA model used in the RG&E PSA. Please describe this model and state how this compares to the Rhodes Model. If different, please discuss the impact of incorporating the Rhodes model into RG&E's PSA. Please identify the type of o-rings currently used in the RCP seals and, if unqualified, please discuss why no SAMA evaluation was provided for this vulnerability.

Response to Clarification 5:

The Ginna Station has two RCPs, each equipped with qualified high-temperature O-rings. The RCP Seal LOCA model used in Revision 4.2 of the Ginna Station PSA is based on a composite from the following:

2. The original Westinghouse RCP Seal LOCA model developed in WCAP-10541, *Reactor Coolant Pump Seal Performance Following a Loss of All AC Power*, Revision 2, November 1986,
3. The RCP Seal LOCA model employed by the NRC in NUREG-1150, *Severe Accident Risks· An Assessment for Five US Nuclear Power Plants*, June 1989, as described in NUREG/CR-4550, *Analysis of Core Damage Frequency From Internal Events: Expert Judgment Elicitation*, Volume 2, Sections 5.4, B-4, and C-4, April 1989;
4 The Rhodes-based Brookhaven National Laboratory model developed in BNL Technical Report W6211-08/99, *Guidance Document for Modeling of RCP Seal Failure*, August 1999,
5 The most recent Westinghouse RCP Seal LOCA model, based on composites from the above, as described in WCAP-15603. *WOG 2000 RCP Seal Leakage Model for Westinghouse PWRs*, Revision 0, December 2000.

This composite is described in Attachment 5.1 to this response, as extracted from Revision 4.2 of the Ginna Station PSA.

Table 1 below compares the probabilities of leakage for the various leakage rates and RCP Seal LOCA models (note that BNL refers to #3 above; also, the values for the Rhodes model have been extracted from Attachment 5.2 to this response). The main difference is that the Rhodes model assumes higher probabilities of leakage in the 57-182 gpm range than the Ginna model (with consequently lower probabilities in the 21-57 gpm range; there is no difference among any of the models for the 182-480 gpm range). Times to core uncovery assuming cooldown for the Ginna and Rhodes models are also shown The Ginna times do not exceed the Rhodes times, and are less in some cases.

To determine the effect on the CDF and LERF from Revision 4.2 of the Ginna Station PSA if the Rhodes model values were substituted for the ones from the Ginna model, a comparison is made in Table 2, showing the substitute values for the three basic events (ACAZSEALX1, -2 and -3) used to model the probabilities of RCP Seal leakage in Revision 4.2. The results of the substitution indicates a slight increase in CDF (0.560%) and no increase in LERF if the Rhodes values are used

Table 1. Probabilities of Leakage and Times to Core Uncovery for Various RCP Seal LOCA Models

<table>
<tr><th rowspan="2">Leakage Rate (gpm/pump)</th><th colspan="5">Probability of Leakage</th><th colspan="2">Time to Core Uncovery (hr)</th></tr>
<tr><th>BNL, NUREG/ CR-4550</th><th>WCAP-15603</th><th>Ginna</th><th>Rhodes</th><th>Ginna-Rhodes</th><th>Ginna</th><th>Rhodes</th></tr>
<tr><td>21</td><td colspan="2">0.7800</td><td rowspan="2">0.8700 (ACAZSEALX3)</td><td>0.7800</td><td rowspan="2">0.7800 (ACAZSEALX3)</td><td rowspan="2">10</td><td>20</td></tr>
<tr><td>21-57</td><td>0.0897</td><td>0.1420</td><td>0</td><td>n/a</td></tr>
<tr><td>57-76</td><td>0.0200</td><td>0.0527</td><td rowspan="2">0.1250 (ACAZSEALX2)</td><td>0.0200</td><td rowspan="2">0.2150 (ACAZSEALX2)</td><td rowspan="2">5</td><td>9.5</td></tr>
<tr><td>76-182</td><td>0.1053</td><td>0.0200</td><td>0.1950</td><td>5</td></tr>
<tr><td>182-480</td><td colspan="5">0.0050 (ACAZSEALX1)</td><td>2</td><td>2</td></tr>
</table>

Table 2. (Import File SAMA_RAI_nu_7.xls)

Table 2. Comparison of CDF and LERF Fussell-Vesely Importances for Ginna vs. Ginna-Rhodes RCP Seal LOCA Models

Ginna PSA Rev. 4.2 with Ginna Probabilities				
Event Name	**Probability**	**CDF Fus Ves**	**LERF Fus Ves**	**Description**
ACAZSEALX1	5.00E-03	0.106%	0.003%	Assumption that loss of seal cooling will fail RCP seals during SBO (480 gpm)
ACAZSEALX2	1.25E-01	2.660%	0.036%	Assumption that loss of seal cooling will fail RCP seals during SBO (< 182 gpm)
ACAZSEALX3	8.70E-01	13.100%	0.248%	Assumption that loss of seal cooling will fail RCP seals during SBO (< 57 gpm)
	Total -->	15.866%	0.286%	
Ginna PSA Rev. 4.2 with Ginna-Rhodes Probabilities				
Event Name	**Probability**	**CDF Fus Ves**	**LERF Fus Ves**	**Description**
ACAZSEALX1	5.00E-03	0.106%	0.003%	Assumption that loss of seal cooling will fail RCP seals during SBO (480 gpm)
ACAZSEALX2	2.15E-01	4.575%	0.061%	Assumption that loss of seal cooling will fail RCP seals during SBO (< 182 gpm)
ACAZSEALX3	7.80E-01	11.745%	0.222%	Assumption that loss of seal cooling will fail RCP seals during SBO (< 57 gpm)
	Total -->	16.426%	0.286%	
Increase with Ginna-Rhodes -->		0.560%	0.000%	

Attachment 5.1

The following has been extracted from Revision 4.2 of the Ginna Station PSA.

4.2.2.3.2 RCP Seal LOCA

In order to limit leakage from the RCS, the RCPs are designed with a seal package comprised of three seals located in series along the pump shaft. This seal package limits RCS leakage by progressively reducing RCS pressure from 2250 psig to containment atmospheric pressure. Failure of this seal package can result in a LOCA similar to any equivalent size pipe break.

The first stage seal in the RCP seal assembly is a "control" film-riding seal which limits leakage along the pump shaft by maintaining a hydrostatic force balance. That is, by maintaining the gap between the non-rotating faceplate of the seal design and the rotating faceplate, leakage can be "controlled." Multiple parameters can affect the gap (i.e., cause it to open or close with the corresponding change in leakage rate) including the angle between the non-rotating and rotating faceplates and inlet fluid pressure and temperature. The second stage seal directs the majority of first stage leakoff to the CVCS system via the seal leakoff line while the third stage seal minimizes the leakage of water and vapor from the pump into the containment atmosphere. These last two seal stages are rubbing type seals with the second stage seal designed to hold RCS system pressure for 24 hours with the first seal leakoff isolated and the RCP static. However, if both the first and second stage seals were to fail, then the third stage seal is not expected to limit leakage (i.e., failure of the first two seal stages is essentially a LOCA).

CVCS is used for the first "control" seal with a portion of CVCS being injected into the primary system and the remainder being directed along the pump shaft and out of the pump via the second stage seal. To provide seal cooling, both CVCS injection into the seal package and seal leakoff from the package must be successful in order to maintain the required hydrostatic balance. In addition to CVCS, component cooling water (CCW) is used to provide cooling water to the RCP motor bearing oil coolers and thermal barrier cooling coil. Per UFSAR Section 5.4.1.1.2 [R.E. Ginna Nuclear Power Plant, *Updated Final Safety Analysis Report*], flow from either CVCS or CCW is sufficient to protect the RCP seals and prevent a possible LOCA. However, if CCW is lost for > 2 minutes, then the RCP must be tripped (if still running) to protect the motor [Ginna Station Procedure AP-CCW.2, *Loss of Component Cooling Water During Power Operation*]. While no specific time limit is provided for loss of both support systems with respect to protecting the seals, it will be assumed that the RCPs must be tripped within the same 2 minutes to prevent seal damage. This 2 minute assumption is also consistent with vendor recommendations per the system engineer.

Westinghouse has extensively studied seal LOCAs during station blackout sequences in which all support systems are lost and the RCPs are not running [WCAP-10541, *Reactor Coolant Pump Seal Performance Following a Loss of All AC Power*, Revision 2, November 1986; and WCAP-15603, WOG*2000 RCP Seal Leakage Model for Westinghouse PWRs*, Revision 0, December 2000]. The two studies are similar in many respects. For example, both include the development of an event tree model to catalog the failure types of seal ruptures and a thermal hydraulic exercise of a code specifically developed to predict seal flow rates. The net result of these studies was that a seal LOCA with a Westinghouse designed pump can result in at most, 480 gpm per pump, or a total flow rate of 960 gpm. This represents the catastrophic failure of all three stages in both RCPs. Using standard conversions (see Tables 4-6 and 4-7), this flow is equivalent to a fixed orifice diameter break of 1.08 inches. The calculation of 1.08 inches is further justified by the fact that a complete severance of a SG tube (0.664 inch inner diameter) results in approximately a 430 gpm leak per the UFSAR which, if ratioed to the tube break area, corresponds

However, the two studies differ in the assumed leakage rates for non-catastrophic failures of the seal package. In the initial study [WCAP-10541], Westinghouse states that leakage through the first stage seal is expected to increase from 3 gpm to approximately 21 gpm per pump if seal cooling is not provided within 30 minutes. Once seal cooling is restored and the seal assembly returns to a thermal equilibrium, leakage will again return to the normal 3 gpm. Without any seal cooling beyond 30 minutes, the seal assembly is vulnerable to failure due to loss of the seal ring geometry (i.e., the seal ring is no longer free to move or "binds") or to degradation of the elastomer material. The Westinghouse analysis predicts a seal failure probability of 2.83E-02 after the first hour of no seal cooling (i.e., there is no failure during the first hour) which gradually increases over time without seal cooling. However, even if there were no failure of the seal assembly, the seals would continue to provide a leak path from the RCS at a rate of 21 gpm per pump until cooling is restored. Westinghouse estimates that core uncovery would occur at approximately 20 hours for a 2-loop plant (see Figure 8-5 of WCAP-10541) under these conditions (i.e., continuous 21 gpm leak per pump).

The more recent Westinghouse analysis [WCAP-15603] predicts seal leakage of 21 gpm after 0 minutes of no seal cooling. After 30 minutes without seal cooling, the seal can catastrophically fail (480 gpm) with a probability of 2.5E-03. The seal can also leak at a higher rate than 21 gpm. That is, after 15 minutes without seal cooling, the seal can continue to leak at a rate of 21 gpm (0.78 probability) or leak at a higher rate up to 480 gpm per pump (0.22 probability).

In addition to the Westinghouse analyses presented in WCAP-10451 and WCAP-15603, an assessment of RCP seal failures is presented in Appendix C.14 of NUREG-1150 [NUREG-1150, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*]. In this analysis, an expert solicitation process (including a Westinghouse representative) was utilized to predict RCP seal leakage rates for various possible seal failure combinations. However, the report states that "prior to 90 minutes, there is no risk of seal failure." Following this 90 minutes, the RCP seal leakage rates and seal failure combinations are generally more conservative than in the Westinghouse analysis.

Finally, a review of the EOPs indicates that operators will isolate all charging flow to the RCPs via needle valves 300A and 300B if all CCW is lost or the RCP #1 seal outlet temperature is offscale high. Further procedure steps isolate CCW to the RCPs (749A/B and 759A/B). This is to prevent the potential for thermally shocking the seals if charging is initiated. Per discussion with operators, this action is expected to occur within 10-15 minutes of loss of charging and CCW. It should be noted that, for a coincident SI and UV condition, all CCW and charging would be lost until CCW is restored in Step 13 of E-0. This is a design condition that also disproves the need to assume instantaneous leakage of 21 gpm with no seal cooling. Normal leakage from an RCP through the #1 seal outlet is 3 gpm.

Based on the above information, the Ginna Station PSA will assume the following with respect to a seal LOCA.

a. Following the loss of all seal cooling (i.e., CVCS and CCW), operators must trip the running RCPs within 2 minutes or a seal LOCA of 480 gpm per pump will result.

b. Following the loss of all seal cooling (and trip of the RCPs), a leakage rate of 3 gpm per pump will be assumed for the first 15 minutes. This is equivalent to 90 gallons (or 10 ft^3) which is only 5% of the water volume in the pressurizer at hot zero power. As such, if seal cooling is restored within 15 minutes so that the leakage rate remains at 3 gpm, no further RCS inventory makeup is required. However, due to concerns with respect to "shocking" the seals following their heatup, the station EOPs required engineering evaluation before restoring seal cooling. Therefore, no recovery of the seals after 15 minutes is credited.

c. Following 15 minutes of no seal cooling, the leakage rate is assumed to increase to 21 gpm per pump, which remains constant until seal cooling is restored or until the seal assembly has failed. Since the operators are procedurally instructed to isolate charging and CCW if seal cooling is lost for >10-15 minutes, the only recovery scenario considered is loss of offsite power. All other equipment failures are assumed to be unrecoverable.

d. Following 15 minutes of no seal cooling, the seal assembly failure probability is dependent on several factors including seal type and whether or not operators have successfully depressurized the RCS. However, for both the WCAP-10451 and NUREG-1150 analyses, seal failure is not postulated until after at least 60 minutes (WCAP-15603 assumes 15 minutes). After this 60 minutes, the seal failure probability increases with time. Assuming a 21 gpm leak per RCP would require RCS makeup at some point in time. Also, a seal failure results in at most 480 gpm per pump, which is the same LOCA size category as the 21 gpm per pump leak per Section 3.4.1.4. Therefore, the only difference between a seal LOCA and maintaining seal integrity after 15-60 minutes is the length of time which is available prior to core damage.

B.4 Seal LOCA Issues

As described in Section 4.2.2.3.2, a reactor coolant pump (RCP) seal LOCA is assumed to result if: (1) operators fail to trip a running RCP within 2 minutes after the loss of all support system cooling (i.e., component cooling water (CCW) and chemical and volume control system (CVCS)), or (2) seal cooling is not restored within 15 minutes following loss of all support system cooling due to long-term seal degradation issues. The first scenario results in a 480 gpm/pump leakage rate while the second results in various leakage rates from 21 gpm to 480 gpm/pump. Since an SBO event can lead to any of the above scenarios, and all leakage rates are within the small-small LOCA category, it must be evaluated whether a seal LOCA falls within the 2.25, 5, or 10 hour offsite power restoration time (i.e., LOCA versus non-LOCA).

For the first scenario, the failure of operators to trip an RCP within 2 minutes upon loss of all seal cooling is defined by event RCHFD00RCP which has a failure probability of 1.61E-02. The failure probabilities for the operators to restore offsite power within 2.25 hours and 10 hours are 0.241 and 0.042, respectively. It is conservative to assume that an SBO induced seal LOCA can use the 10 hour non-LOCA offsite restoration time since:

f_{SBO} * RCHFD00RCP * 2.25 Hour Restoration < f_{SBO} * 10 Hour Restoration
f_{SBO} * (1.61E-02) * (0.241) < f_{SBO} * (4.2E-02)

For the second scenario, various publications exist with respect to the potential for RCP seal failures [WCAP-10541, NUREG/CR-4550, BNL Technical Report W6211-08/99 and WCAP-15603]. For the purposes of the Ginna Station PSA analysis, the probabilities presented in BNL Technical Report W6211-08/99, along with data from NUREG/CR-4550 will be used since they are currently accepted by the NRC. Specifically, the BNL report and WCAP-15603 identify probabilities of seal leakage per RCP with qualified high-temperature O-rings, as installed at Ginna, as follows:

Leakage Rate	BNL, NUREG/CR-4550	WCAP-15603
21 gpm	0.78	0.78
21 - 57 gpm	0.0897	0.142
57 - 76 gpm	0.02	0.0527

76-182 gpm	0.1053	0.02
182 - 480 gpm	0.005	0.005

It is assumed that offsite power must be restored within 1 hour if the TDAFW pump is failed or up to 10 hours if it is available and providing RCS cooldown. The one-hour TDAFW pump limit can be assumed to identify whether or not RCS cooldown is occurring. We will initially assume that RCS cooldown always occurs.

Using Table 2-3 from the BNL report, various times are calculated for core uncovery with different seal LOCA sizes and timing. Multiplying the leakage rates above by the number of RCPs at Ginna Station (i.e., two), yields the following times for core uncovery:

42 gpm total >17 hours
114 gpm total >17 hours
152 gpm total >17 hours
364 gpm total ~ 8 hours
960 gpm total ~ 5 hours

It should be recognized that the above times are for a 4 loop Westinghouse plant which has both a larger RCS inventory and more fuel (i.e., decay heat). To account for these differences, the following will be assumed for the Ginna Station PSA:

0.870 probability that a seal LOCA between 21 and 57 gpm will occur (basic event ACAZSEALX3). This break size will use the 10 hour recovery time per the above discussion.

0.125 probability that a seal LOCA between 57 and 182 gpm will occur (basic event ACAZSEALX2). This break size will use the 5 hour recovery.

0.005 probability that a seal LOCA between 182 and 480 gpm will occur (basic event ACAZSEALX1). This break size will use the 2 hour recovery.

It should be noted that the above times also bound those in the BNL report and NUREG/CR-4550 for cases where the RCS is not cooled down. Therefore, the previous assumptions remain valid.

Attachment 5.2

The following has been extracted from Attachment A, "The USA NRC RAIs on WCAP-15603, Rev. 0, and WOG Responses to the RAIs," in WCAP-15603, *WOG2000 RCP Seal Leakage Model for Westinghouse PWRs*, Rev. 1, May 2002. It provides the quantitative description of the Rhodes model used for the comparison in this response.

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15603

Revision 1

WOG 2000
Reactor Coolant Pump Seal Leakage Model
for
Westinghouse PWRs

Selim Sancaktar
Reliability and Risk Assessment

May 2002

Reviewer: ______________________________
John Kitzmiller
Reliability and Risk Assessment

Approved: ______________________________
Jim Brennan, Manager
Reliability and Risk Assessment

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ATTACHMENT A

THE USA NRC RAIS ON WCAP-15603, REVISION 0

AND

WOG RESPONSES TO THE RAIS

WESTINGHOUSE OWNERS GROUP
WOG 2000

REACTOR COOLANT PUMP SEAL LEAKAGE MODEL FOR WESTINGHOUSE PWRS

WCAP-15603, REVISION 0

RESPONSES TO THE NRC RAIS

The answers to the 10 RAIs are given below. WCAP-15603 is revised to leave out the RCP seal leakage models for unqualified o-rings to minimize the points of contention. Almost all domestic Westinghouse nuclear power plants have either switched to qualified o-rings, or are scheduled to do so in the near future. Since the RCP seal leakage model for the old o-rings is more involved, we propose not to address it in WCAP-15603 – i.e., to remove reference to old o-rings from WCAP-15603.

NRC REQUEST FOR ADDITIONAL INFORMATION

RAI 1 The Topical Report states in Section 1.0 (page 1-1) that the Brookhaven National Laboratory (BNL) model is the current regulatory model for reactor coolant pump (RCP) seal leakage, and it uses this model as the starting point for the development of the WOG 2000 model. However, the BNL model is not the current regulatory model. The staff committed in resolving Generic Issue 23 to use the Rhodes model until other acceptable reactor coolant pump (RCP) seal models were developed. The original intent of the BNL report was to interpret and clarify the other existing RCP seal models, including the Rhodes model. However, as part of their report, BNL developed their own best-estimate RCP seal model, which differed from the other seal models. In developing the BNL best-estimate model, BNL made assumptions regarding seal failure with which the U.S. Nuclear Regulatory Commission (NRC) staff may not fully agree. For example, the Brookhaven model uses a probability of 0.54 for the popping-and-binding failure mode for the third-stage seal, given that the second stage seal has failed, and the WOG 2000 model reduces this probability to 0.27 for the "new" o-rings. However, the Rhodes model assumed pop-open failure of the third stage seal under these conditions (i.e., probability of one). The Topical Report needs to address and justify the differences between the WOG 2000 model and the Rhodes model.

Response to RAI 1:

The sentence mentioned in the RAI will be modified in WCAP-15603 to describe the BNL Report as the "primary reference" for the WOG2000 model.

The basis for considering the BNL Report as the primary reference is that (a) it contains a best estimate RCP seal Leakage PRA model and (b) it represents the most current effort to provide a reasonable middle ground for this expert-opinion-driven issue. In our opinion, the 'Rhodes Model' represents one opinion, a conservative one at that, which may have been adequate for generically addressing the USI 23 (the purpose of NUREG /CR-4906P and NUREG/CR-5167), but is not appropriate for plant PRA modeling and decision making. Foundational to PRA

philosophy is that the plant risk should not be distorted by conservative assumptions which would mis-focus the component and system importances and PRA insights for risk-informed applications. This position has been supported both by utilities and by NRC. Moreover, it is our opinion that the "Rhodes Model" has not undergone the test of adequate PRA modeling compatible with the current PRA philosophy and practice.

The whole point of this WCAP is NOT to provide new evidence or analysis to re-evaluate the seal leakage phenomena BUT to agree upon a mutually acceptable PRA model, in a 15-year-old, expert-opinion-driven issue. For this purpose, the WOG2000 model refers to the NRC-sponsored BNL report and uses the best estimate model in that report in the spirit of recent PRA practices and philosophy shared by the NRC and the utilities. The differences between the Brookhaven and WOG2000 leakage models are clearly identified and discussed in WCAP-15603. These differences are introduced to have a realistic representation of the phenomena involved. We have no new analysis or tests to provide.

However, to be responsive to the RAI, we compared the WOG2000 leakage model with the latest discussion we could find of the Rhodes model, namely in Appendix A of NUREG/CR-5167 (April 1991). The discussion in NUREG/CR5167 references to NUREG/CR-4906P (January 1988), but the probabilities and times between the 2 versions differ.

Areas to be compared:

1. *Failure probability for first stage*

   The Rhodes PRA Model in NUREG/CR-5167 gives the failure probability for the first stage as .025. The WOG2000 model gives the failure probability for the first stage as .0125. The rationale for this failure probability is provided in WCAP-15603. The BNL PRA model gives the failure probability for the first stage as .025.

2. *Failure probability for second stage*

   Assumptions are the same across all 3 models.

3. *Failure probability for third stage*

   The Rhodes PRA Model in NUREG/CR-5167 gives the failure probability for the third stage as 1.0. The WOG2000 model gives the failure probability for the first stage as 0.27. The rationale for this failure probability is provided in WCAP-15603. The BNL PRA model gives the failure probability for the first stage as .54.

4. *Seal leakage rates for failures*

   Assumptions are the same across all 3 models.

5. *O-ring failure probability as a function of time after start of event*

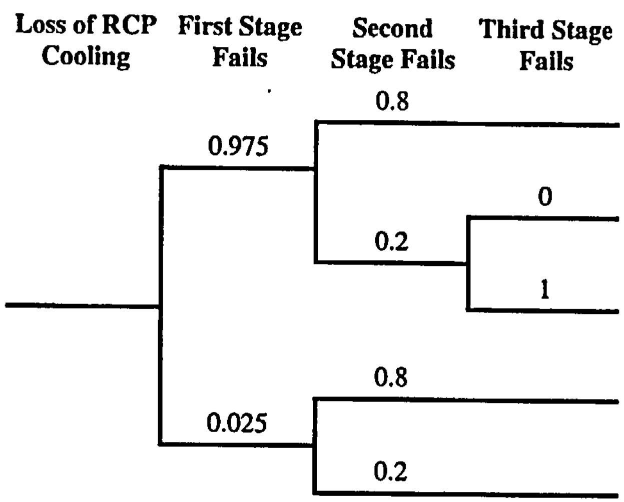
NUREG/CR-5167 on page A-6 states that (based on Section 8 of WCAP-10541) the plant cooldown is assumed to have little effect on the assumed failure time of the o-rings (which are assumed to fail after 2 hours.) The Rhodes model distinguishes between "Improved" o-rings qualified by Westinghouse and "Qualified" o-rings which would be qualified to withstand 550°F and 2250 psi. However, Section 8 of WCAP-10541 indicates that RCS pressure would be reduced starting almost immediately, either through loss of RCS inventory or through operator action to cool the plant and depressurize.

The material presented in WCAP-10541 Section 8, Figure 8-3 indicates that the leakage from the seals results in a slowly decreasing pressure in the RCS for the case of 15 gpm leakage per pump such that RCS pressure is < 2000 psi at 1 hour and ~ 1800 psi when the cooldown is started and < 1600 psi at 2 hours. As noted in the WCAP–10541 text, cooldown was assumed to start (for purposes of these analyses) when pressurizer level decreased to < 10%. For the case with 300 gpm leakage per pump (Figure 8-1) the cooldown started at approximately 15 minutes at an RCS pressure of ~ 1600 psi and reached a pressure of < 1000 psi in approximately 30 minutes. Test and qualification data for high temperature o-rings indicates that all o-ring and gap combinations tested (120 o-rings were tested in the original qualification testing and 188 o-rings have been tested in supplemental batch testing for a total of 308 o-rings tested) did not fail during the 18 or 168 hour test period and the absolute minimum pressurization failure pressure for any combination was 1710 psi. Based on this information, assuming failure at 2 hours (as is stated on page A-6 of NUREG/CR-5167) is overly conservative. Additional information on o-ring testing results is provided in the response to RAI-6.

The comparisons (Rhodes, BNL report, and WOG2000) are provided in Figures A-1 through A-3.

A new data point for seal behavior has been established with the Maanshan SBO event (March 2001), in which the seals were exposed to hot standby RCS pressure and temperatures, which lasted for two hours, with no indications of excessive seal leakage. The seals in one RCP were inspected after the event and were found to be in good condition. Based on this inspection, the seals in the other 2 RCPs were not inspected and were continued in service for the remainder of the operating cycle.

Figure A-1 Our Understanding of the Rhodes Model
(Qualified O-Rings Best Estimate)



Scenario	Leakage (per pump) (From NUREG/ CR-4906)	Probability	Start of Core Uncovery (hours) With Cooldown	Start of Core Uncovery (hours) Without Cooldown	Comments
1	21	0.78	20	16	Scenario probabilities and core uncovery times are inferred.
2	57	0			
3	182	0.195	5	3.5	
4	76	0.02	9.5	7	
5	480	0.005	2	1	2 means that the core uncovery probability is zero from 0 to 120 minutes and jumps up to 0.005 immediately at 120+th minute.
Sum =		1			

Scenario start time = Not stated. Last paragraph on page A-1 of NUREG/CR-5167 states that the first stage inlet temperature would reach prevailing RCS temperature in 10+3 = 13 minutes, per Westinghouse predictions.

O-ring failure probability = 50% after 2 hours after full system delta P is applied. It is not explained how this is used in the scenarios. It is also not explained what this probability is when the RCS temperature/pressure is reduced first by reactor trip and AFW system operation; next by operator rapid cooldown per ERGs.