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DCP/NRC0475
Docket No.: STN-52-003

March 8, 1996

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

ATTENTION: T. R. QUAY

SUBJECT: WESTINGHOUSE RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION ON THE AP600

Dear Mr. Quay:

Enclosed are the Westinghouse responses to NRC requests for additional information on the AP600 Design Certification program. Enclosure 1 contains responses to 10 follow-on questions pertaining to Level 1 PRA topics. Enclosure 2 contains responses to two follow-on questions pertaining to accident management.

Based on WCAP-13913, which provides the framework for the AP600 severe accident management guidance, and the response to the two follow-on questions provided in Enclosure 2, Westinghouse believes that DSER open item 19.2.5-1 pertaining to accident management is closed.

A listing of the NRC requests for additional information responded to in this letter is contained in Attachment A.

Please contact Cynthia L. Haag on (412) 374-4277 if you have any questions concerning this transmittal.

John C. Butler for BAM
Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/nja

Enclosure
Attachment

cc: D. Jackson, NRC (1 copy enclosure)
R. Palla, NRC (w/o enclosure/attachment)
J. Flack, NRC (w/p enclosure/attachment)

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Enclosure 1 to Westinghouse
Letter NSD-NRC-96-4662

March 8, 1996

2703A

RESPONSES TO DSER 19.1 (PRA) OPEN ITEM FOLLOW-ON QUESTIONS

RAI Related to DSER Open Item 19.1.3.1-1

1. The Passive Residual Heat Removal (PRHR) tube rupture frequency was chosen by Westinghouse to be 5.0E-4/year on the basis that it should be approximately an order of magnitude lower than the frequency of a Steam Generator Tube Rupture (SGTR) event. If Westinghouse's approach, based on a pipe break failure rate of 4.25E-10 per year per section, was followed, this frequency would be 5.0E-3/year. If the failure rate for PRHR heat exchangers of 1.0E-7/year (recommended by EPRI in its Utility Requirements Document) were used, the PRHR tube rupture frequency would be 2.0E-3/year.

Please re-evaluate the PRHR tube rupture frequency by taking into account the following: (1) it is not possible to isolate and repair a single leaking PRHR heat exchanger without a plant shutdown, (2) the possibility of stress corrosion which accelerates under stagnant conditions by allowing local concentrations of ions or oxygen, (3) the efficiency of detecting very small leaks to a very large body of water (in the IRWST), under stagnant conditions, may not be better than the leak detection capability of circulating primary in the steam generators, (4) the potential impact of mechanical loads on heat exchanger tubing and supports, including potential steam hammer load caused by phase separation within the tubes under accident conditions, and (5) the smaller heat transfer area of PRHR heat exchanger, as compared to steam generators, combined with the potential for two-phase flow in the IRWST side of the tubes during accident conditions where critical heat flux and vapor blanketing of the tubes may be of concern.

Response:

1. There are a number of factors that were considered in determining the PRHR heat exchanger (HX) tube rupture frequency used in the AP600 PRA. Westinghouse has related the PRHR HX tube rupture frequency to more recent Westinghouse steam generator data because of similarities in design features and extensive operating experience. The PRHR HX design is not similar to a piping system or to a normal residual heat removal heat exchanger. The reasons for selecting this initiating event frequency are discussed in PRA Subsection 2.3.1.3. The considerations used in determining the tube rupture frequency include:

   a. The PRHR HX utilizes materials and design features that are very similar to the AP600 steam generator. The tubes are the same material. The PRHR HX tube to tubesheet joint is made the same way as in the AP600 steam generator. In addition, there are 94% fewer tubes in the PRHR HX as compared to the AP600 steam generators (671 vs 12,614). Each PRHR HX tube is shorter (avg length of ~33 ft vs ~66 ft for the steam generator) and has fewer support interfaces (7 vs 24 for the SG).

   b. The PRHR HX has a reduced chance of sludge buildup because the tube sheet is vertical and the HX is not normally operating. These design features reduce the chance of corrosion around the tube sheet joint.

   c. The PRHR HX will see stagnant conditions on the primary side, but with reactor quality water (very low oxygen concentrations). The IRWST side will see oxygenated water, but is not really stagnant. In addition, the PRHR HX normally sees low temperatures (< 120 F). These fluid conditions reduce the chance of primary side corrosion.

d. The PRHR HX normally sees no flow. This condition significantly reduces the chance of vibration related failure mechanisms.

e. PRHR HX leak detection is provided by several features, including an RTD located in the cold trapped part of the PRHR HX inlet line, a pressure transmitter located between the PRHR HX isolation valves and the other RCS leak detection features. It is estimated that the inlet line RTD can detect leaks of about 0.03 gpm. The PRHR HX pressure transmitter can determine if a PRHR HX leak is inside or outside the isolation valves. RCS leak detection instruments (containment sump level, containment radiation, and RCS mass balance) are used to quantify the leak.

   AP600 does not rely on the IRWST level or temperature instruments to identify PRHR HX leakage.

f. The PRHR HX tubes do not see two phase conditions inside the tubes during non-LOCA accidents. Steam enters the PRHR HX only during some low probability condition 3 or 4 events. In addition, steam condensing inside the tubes does not lead to large loads on the tubes because of the geometry of the tubes and the fluid conditions inside the tubes (hot water / steam). Water hammer was not observed in the PRHR HX during AP600 testing. Therefore, steam condensation loads are not an issue for PRHR HX tube rupture frequency.

g. Critical heat flux was not observed during the PRHR HXs test and is not expected to occur in the plant. Even if it did occur, it does not represent a threat to tube integrity; the heat source is reactor coolant water or steam which can not overheat the tube like a nuclear fuel rod or an electrical heater rod.

h. As a result of all these features and conditions, the PRHR HX will also have a low probability of a tube leak. The Technical Specifications will allow plant operation with a small PRHR HX leak and will require that the plant be shutdown before a PRHR HX leak could degrade into a tube rupture.

Based on this evaluation, the PRHR HX tube rupture frequency selected for the AP600 PRA is conservative.

RAIs Related to DSER Open Item 19.1.3.1-1

2. The primary system pipe break analysis assumes a certain apportionment of the failure rate, according to pipe sizes, into "large", "medium", "intermediate " and "small" LOCAs. Although such apportionment is logical, the assumed percentages are rather arbitrary. Sensitivity analyses are needed to assess the impact of this apportionment on the PRA results and insights.

Response:

The random pipe leakages severe enough to constitute a rupture are classified in the AP600 PRA in accordance with the ALWR URD. These classifications are:

10% complete break

30% large rupture

60% small rupture

This is the breakdown used in the Oconee PRA, which the NRC reviewers found to be acceptable.

To see how a variation in the random pipe rupture frequency would affect the core damage frequency, consider the following:

In Table 2-4 of the AP600 PRA, the initiating event (IEV) frequencies (based upon the above random rupture apportionments), and contributions of the various relevant LOCA sizes to the plant core damage frequency are:

Table 1

LOCA Category	IEV Frequency (per year) [F]	Core Damage Frequency (per year) [CDF]	Conditional Core Damage Probability [CDF/F]
LLOCA (D>= 9")	1.1E-04	2.6E-08	2.4E-04
MLOCA (5< =D<9'')	1.6E-04	5.0E-09	3.1E-05
NLOCA (2< =D<5")	7.7E-04	3.0E-08	3.9E-05
Totals	1.0E-03	6.1E-08	N/A

To see how the AP600 PRA core damage frequency would change if the random rupture apportionments for LOCA were to be changed, use the following variations in apportionments in random rupture frequency. To simplify the discussion the IEVs shown in Tables 2 and 3 include only the contribution from the random pipe rupture. The other contributors discussed in Table 2-4 of the AP600 PRA are included in Table 4 of this discussion.

For large piping:

Table 2

LOCA Category	Current Apportionment of Frequency	Current IEV[1]	Variation (A)	Variation (A) IEV Freq[1]	Variation (B)	Variation (B) IEV Freq[1]
LLOCA	10%	5.1E-5	20%	1.0E-4	5%	2.6E-5
MLOCA	30%	1.6E-4	40%	2.1E-4	25%	1.3E-4
NLOCA	60%	3.8E-4	40%	2.1E-4	70%	3.6E-4

For medium piping:

Table 3

LOCA Category	Current Apportionment of Frequency	Current IEV[1]	Variation (A)	Variation (A) IEV Freq[1]	Variation (B)	Variation (B) IEV Freq[1]
MLOCA	40%	7.45E-6	60%	1.1E-5	30%	5.6E-6
NLOCA	60%	1.12E-5	40%	7.4E-6	70%	1.3E-5

Notes:

1) The IEV frequencies shown here include only pipe rupture They do not include the other factors shown in Table 2-4 of the AP600 PRA which are used to get the IEV frequency used to quantify the AP600 PRA.

The new core damage frequency contributions from LOCA that would result from these two variations in apportionment of the LOCA sizes are:

Table 4

LOCA Category	Current CDF	Variation (A) IEV Freq	Variation (A) CDF[2]	Variation (B) IEV Freq	Variation (B) CDF[2]
LLOCA	2.6E-8	1.6E-4	3.8E-8	8.1E-5	1.9E-8
MLOCA	5.0E-9	2.2E-4	6.9E-9	1.4E-4	4.2E-9
NLOCA	3.0E-8	6.1E-4	2.4E-8	7.7E-4	3.0E-8

Notes:

2) The IEV and CDF in this table include all of the factors shown in Table 2-4 of the AP600 PRA that go into the quantification of the PRA.

The total contribution to the CDF from LOCA using the apportionments of Variation(A) is 6.9E-8. The total contribution to the CDF from LOCA using the apportionments of Variation(B) is 5.3E-8. The total contribution to the CDF from LOCA in the AP600 PRA is 6.1E-8. Thus, there is little sensitivity in the CDF from LOCA to variations in the pipe break apportionments.

RAIs Related to DSER Open Item 19.1.3.1-1

3. The next PRA revision should reflect the PRHR design change, i.e., one instead of two heat exchangers.

Response:

One PRHR heat exchanger will be reflected in the AP600 PRA update.

RAIs Related to DSER Open Item 19.1.3.1-2

1. Westinghouse is requesting the extension of the testing interval, from quarterly to semi-annually, for the ADS stage 1, 2 and 3 motor operated valves (MOVs). The FSAR states in 3.9.6.3.1 that the ADS stage 1 through 3 valve exercise testing represents a risk of loss of coolant and depressurization of the reactor coolant system if the test sequence is not followed. Operator error during exercise testing of ADS MOVs (e.g., failure to follow test sequence) must be addressed in the PRA.

Response:

Potential operator error during testing of ADS stage 1, 2, and 3 MOVs, leading to spurious ADS actuation will be addressed in the spurious ADS actuation initiating event, in the next PRA update.

RAIs Related to DSER Open Item 19.1.3.1-2

2. A methodology is given in Section 26.5.3 for calculating the frequency of spurious ADS actuation from a 2 out of 2 signal train. Section 11.1.2, however, indicates that ADS actuation is based on 2 out of 4 level detectors in either of the 2 CMTs, which includes 12 possible combinations of 2 signals. The staff was unable to find in the revised PRA submittal a description of the analysis with enough details to understand how the contributions to intermediate, medium and large LOCA, reported in Section 3.5.3, were calculated. Please provide a clear description of the analysis (including assumptions, data and associated bases) used to calculate ADS spurious actuation frequencies and their contributions to the various LOCA initiating event frequencies.

Response:

The spurious ADS signal was modeled by the ADS-IC83 fault tree and uses 2-out-of-4 logic. Please refer to this fault tree for the failure logic used.

The frequencies of large, medium, and intermediate LOCA events for spurious ADS actuation are modeled in Chapter 3 of the AP600 PRA, by the fault trees named ADLF, ADMF, and ADNF. Please refer to these fault trees to see the failure modes modeled. These contributions are also discussed in Subsection 2.3.1.7 of the PRA.

RAIs Related to DSER Open Items 19.1.3.1-4 and 19.1.3.1-6

1. Westinghouse assumes a mission time of 24 hours for long-term cooling independently of plant condition. This assumption must be justified by showing (e.g., through a bounding analysis) that the remainder risk (beyond 24 hours) is not significant. Otherwise the event tree models must be extended beyond 24 hours (to a point in time where it can be argued that the remainder risk is not significant).

Response:

Each core damage event tree is modeled through a time frame of 24 hours. This time frame is selected to be consistent with the commonly accepted PRA practices. It is postulated that in such a time frame, an event sequence will reach steady state, stable conditions which will change only very slowly at the end of that time frame. Further, there exists the possibility of numerous and diverse recovery actions (which are not credited in the accident sequences) that can be undertaken in such a time frame to further mitigate the event.

Another important aspect of this time frame is that it covers the actuation of all expected mitigation functions/systems needed to avoid core damage.

In general, if during the first 24 hours following event initiation the reactor achieves a stable shutdown condition without core damage, as the result of successful system functions and/or operator actions, and without the need for further action or system operation, the sequence is categorized as successful. Core damage is assumed if the RCS conditions are not stabilized in 24 hours, or if core damage is anticipated following 24 hours without further system or operator action.

The core damage risk at shutdown conditions is already covered by the AP600 shutdown risk model. Thus, the remaining potential question is the following: Is there a residual core damage or severe release risk during the time interval between the times postulated in the at-power and shutdown PRA models?.

The residual risk during the above time interval between at-power and shutdown models is judged to be negligibly small for the following reasons:

a. For those accident sequences that end with successful recirculation after IRWST injection, the failure probability of this passive function after successful actuation is very small; moreover, RNS pumps are a backup to this system.

b. For those accident sequences that end with successful heat removal by RNS, the remainder of risk is accounted for in the shutdown model, where loss of RNS is modeled during shutdown.

c. For those transient sequences that end with success of a secondary heat removal system (MFW or SFW), the remainder of the risk is negligible during transition to shutdown, since credit can be taken for multiple passive safety systems. This risk is bounded by the risk due to transients, which is already small.

d. For the functions of interest (e.g., gravity injection, normal RHR, passive RHR, startup or main feedwater), the fault tree mission times are 24 hours after system actuation, not after event initiation. Thus, many event sequences, particularly for such events as transients and small LOCAs, effectively consider a period longer than 24 hours.

e. It has been established that adequate containment cooling can be provided solely by air cooling of the containment surface. Thus, no other containment function is necessary, although such systems (such as passive containment cooling or containment fans) will most likely be available.

f. AP600 passive safety systems, once actuated, are not susceptible to the "run" failures associated with active safety systems. Since actuation faults for all of the passive safety systems are modeled in the event during the first 24 hours, the residual risk of failure of these systems is much less in any following time interval. Thus, the residual risk after 24 hours for AP600 is expected to be much less than that of current PWRs.

For the above reasons, it is concluded that the mission time of 24 hours is appropriate, since any residual risk resulting from conditions that might occur between the times modeled in the at-power and the shutdown PRA models would be small.

RAIs Related to DSER Open Item 19.1.3.1-7

1. Westinghouse needs to correct several inconsistencies or provide an explanation indicating that the apparent inconsistency resulted from a misunderstanding. Several entries in the "System Dependency Matrix" tables, at the end of each specific system chapter, are inconsistent with the "AP600 Support System Interdependency Matrix" table located in Chapter 5. Examples are:

- For the Passive Containment Cooling System (PCS), Table 13-4 on page 13-9 of the PRA, shows that IDS is the support system required to operate AOVs and MOVs. However, PCS-PCT in Table 5-6 on page 5-30 of the PRA does not show that the IDS system is a support system.

- For the Normal Residual Heat Removal System (RNS), Table 17-4 on page 17-10 states that PLS system provides manual actuation logic for pumps, MOVs, etc. However, RNS-RHR and RNS-RNP (Table 5-6 on page 5-33) indicate the PMS system (not the PLS system) provides support.

In addition, Section 21.4.2 refers to subsection 8.3.1 of reference 21-1. Reference 21-1 is the revision 1 fault trees and there is no subsection 8.3.1. The correct reference should be given.

Response:

o Chapter 5 dependency matrices represent what is modeled in the fault trees. The dependency matrices in system sections may show all formally existing support systems, regardless of whether they are used in a given mission. For example, in PCS, the MOVs and AOVs are supported by IDS. But, the MOVs are normally open, and the AOVs fail open if IDS fails. Thus, IDS does not show up in the fault trees.

- For the RNS, the I&C modules will be updated, if necessary. The valves are actuated by PMS, and the pumps are actuated by PLS. Thus, both the PMS and PLS are support systems for the RNS.

- The reference 21-1 in Chapter 21 is a typographical error. It should be replaced by "AP600 Standard Safety Analysis Report".

RAIs Related to DSER Open Item 19.1.3.1-10

1. The staff requested Westinghouse to assess and document the applicability of generic failure data to the AP600 design. While check valves are not unique to the AP600, the conditions under which they will be operating in the plant are substantially different from those in current generation nuclear plants. For example, they will have to open on demand under very low differential pressures after long periods of being held closed by fluid at RCS temperature, pressure and chemistry. In the revised PRA submittal the failure rate of the IRWST check valves was changed, as suggested in EPRI's Utility Requirement Document, to account for "less than ideal conditions" which may exist at the time the valves are demanded. However, no discussion is included in the submittal which shows that this change addresses the failure data applicability concern for the IRWST check valves or for any other components. Please provide this information and/or perform sensitivity studies to assess the impact of changes in failure rates of risk-important components to risk.

Response:

The only component type that might be subject to this "low differential pressure" concern is check valves in passive systems. This was confirmed in a telephone conversation with the NRC PRA staff on Jan. 22, 1996. Since the other types of valves (MOVs, AOVs, squib valves, etc.) in the AP600 passive systems operate under the same types of conditions as in current plants, and no different failure modes exist for these components, there is no new failure mode such as a low differential pressure to consider for these components.

Some of the check valves in the IRWST system design have been replaced with squib valves. This reduces the number of check valves and eliminates the high differential pressure normal operating environment that the valves in the IRWST injection and recirculation lines would experience in the previous design. Thus, the AP600 check valves now have operating conditions within the range of applicability of the generic check valve failure data.

RAIs Related to DSER Open Item 19.1.3.1-13

1. In calculating the common cause failure (CCF) probability of the IRWST injection line check valves, MGL factors from Revisions 5 and 6 of EPRI's Utility Requirements Document (URD) were used. A beta factor of 0.026 is recommended in Revisions 5 & 6 of the URD. This is much lower than the value recommended in previous revisions of the URD (i.e., 0.17) as well as in previous PRAs (e.g., System 80+). No explanation for this is provided. Please explain the reasons for assuming a value for such beta factor which is considerably lower than values used in previous PRAs.

Response:

The estimation of the MGL common cause parameters for check valves is given in the ALWR URD document, based on 22 independent events in the EPRI data bank. In that data there are no CCF events involving failures of check valves to open. The following discussion on the history of the estimates is taken from a March 8, 1991 SAROS letter.

" *When the common cause parameters were generated for the Key Assumptions and Groundrules (KAG) document for the passive ALWRs, they were based almost entirely on the methods and data base compiled by EPRI. The data base [originally] available did not, however, cover check valves. To obtain estimates of the [MGL] parameters for check valves, which are relatively more important for passive plants than for evolutionary and current-generation plants, the event summaries for LERs included in NUREG/CR-1363 were reviewed. These summaries are not very detailed, but they appeared to be the best source of information available.*

More recently, EPRI has updated the data base on common cause failures. This updated data base now covers check valves. [This update] ... performed a much more careful examination of individual events than was done for NUREG/CR-1363, or that is possible based on the event summaries in that report. Moreover, some of the events that were important in the assessment based on NUREG/CR-1363 are not reported elsewhere, including in the revision to NUREG/CR-1363, giving rise to the suspicion that they were not characterized in the original NUREG as well as they might have been.

A different look has therefore been taken at the common cause parameters for check valves, based on the new EPRI data base. In particular, the failure of check valves to open has been recalculated. Because check valves are generally quite reliable when they are required to open, there is the problem that the data base on failures is relatively limited. EPRI found no common cause failures to open of check valves (other than failure modes unique to testable check valves). The updated common cause assessment is therefore based on evidence of no common cause failures, with a data base that included 22 independent failures to open. ... A one-third failure (in this case, a non-lethal failure of two check valves to open) [has been assigned] as representative of the mean of the actual failure rate when no failures have been experienced (instead of calculating an upper bound)." The resulting evaluation is given in the ALWR URD on page A.A-117.

The assignment of 0.33 common cause failures per 22 random failures in the URD results in a beta factor of 0.029 for CCF of check valves to open. This can be favorably compared to a beta factor determined from a Bayesian update of a generic prior distribution. Assume a beta factor based on a generic prior mean of 0.1 with an error factor of 10 for the check valves. Update this value with the evidence of 0 CCF in 22 random failures, assuming a lognormal distribution. This Bayesian update produces a mean beta factor of 0.023, which is less than the one reported in the URD. If a more conservative prior mean of 0.2 with an error factor of 10 is used, the posterior mean beta factor

becomes 0.031, which almost the same as the one reported in the URD. This provides a basis for the beta factor reported in the URD.

RAI Related to DSER Open Item 19.1.3.1-16

1. It is not clear whether unscheduled maintenance that could affect the unavailability of safety-related "passive" systems is modeled in the PRA. For example, it is mentioned (Table 9-5) that the normally closed air-operated valves in the Core Makeup Tanks are exercise-tested every three months. Unavailabilities are based on quarterly testing, which implies that faulty valves will be repaired upon detection, the valve unavailability due to such unscheduled maintenance is not modeled in the PRA (neither a justification for not modeling it is provided). This seems true, also, for several other systems, such as the PRHR and the ADS. Please address unscheduled maintenance in the PRA.

Response:

CMT air-operated valves are tested quarterly. Unavailability of a CMT train due to unscheduled maintenance is mentioned in the system modeling; but the corresponding basic event was inadvertently left out of the fault tree models. This will be corrected in the PRA revision.

PRHR valves are tested quarterly. The test and maintenance unavailability of PRHR valves PXS-V108A and B are modeled in the fault trees by the basic events PRBAV108TM and PRBEV108TM.

ADS motor-operated valves are tested every six months. If a valve is found to be inoperable, the action statement associated with its tech spec is implemented. Failures due to valve actuation can be repaired; failures associated with the valve body (RCS pressure boundary) can not be fixed and require a plant shutdown. Given that most likely failure modes of the motor-operated valves would be associated with the actuator, rather than the valve body, unscheduled maintenance of ADS valves during power operation is possible. This will be addressed in the fault trees of the PRA revision.

Enclosure 2 to Westinghouse
Letter NSD-NRC-96-4662

March 8, 1996

Response to Follow-on Questions Related to DSER OI 19.2.5-1

480.212 Identify and discuss actions that would be required to prevent or mitigate uncontrolled fission product releases after 24 hours due to (a) long term non-condensible gas generation, (b) depletion of coolant inventory due to normal leakage and early bypass sequences, (c) late containment bypass (temperature-induced SGTR), and (d) depletion of PCSS water inventory.

Response: As discussed in the response to RAI 720.55 and RAI 720.56, Westinghouse has developed a framework and a set of high level strategies for severe accident management. This work is documented in "Framework for AP600 Severe Accident Management Guidance", WCAP-13913, December 1993. High level strategies to diagnose potential fission product release pathways and then to prevent, terminate and/or mitigate those fission product releases are identified and discussed in WCAP-13913. The high level strategies presented WCAP-13913 are applicable to all of the items outlined in this question.

Westinghouse believes that the development of the framework for a severe accident management program for the AP600 plant design, including the identification of high level strategies provides a sufficient basis for the development of the detailed AP600 Severe Accident Management Guidance by the COL applicant.

480.213 Provide an assessment of how the following accident management related strategies could be implemented, and the degree to which these strategies would be facilitated by either AP600 design features or pre-planning to be carried out by either the vendor or the COL-applicant:

a. bypassing emergency trip for NRHR and CVCS pumps

b. bypassing the emergency trip for the DGs

c. bypassing the MSIV closure signals

d. backing up the plant air system with nitrogen bottles

e. providing coolant to the containment shell in the event of either PCSS hardware failure or inventory depletion

f. recharging the station batteries using portable battery chargers

Response: None of the accident management strategies described above were credited in either the design analyses or the PRA analyses for the AP600 design. Since their contribution to preventing or mitigating core damage accidents is not significant, special design features or pre-planning to accommodate the use of these strategies are not appropriate. However, the need for development of detailed steps required to implement these strategies, such as bypassing trips, will be considered during the development of the detailed severe accident management guidance for the AP600 design. As discussed in the response to Q480.212, the development of detailed severe accident management guidance will be completed by the COL applicant prior to initial operation of the plant.

Attachment A to NSD-NRC-96-4662
Enclosed Responses to NRC Requests for Additional Information

Re: Level 1 PRA

RAIs 1, 2 & 3 as related to DSER OI 19.1.3.1-1
RAIs 1& 2 as related to DSER OI 19.1.3.1-2
RAI 1 as related to DSER OIs 19.1.3.1-4 & 19.1.3.1-6
RAI 1 as related to DSER OI 19.1.3.1-7
RAI 1 as related to DSER OI 19.1.3.1-10
RAI 1 as related to DSER OI 19.1.3.1-13
RAI 1 as related to DSER OI 19.1.3.1-16

Re: Accident Management (DSER OI 19.2.5-1)

480.212
480.213