REQUEST FOR ADDITIONAL INFORMATION 423-2710 REVISION 1

7/9/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: PRA

QUESTIONS for PRA Licensing, Operations Support and Maintenance Branch 1 (AP1000/EPR Projects) (SPLA)

19-359

In the response to RAI Question 19-83 and in reference to two operator actions (i.e., depressurization by secondary side cooling and switchover from containment spray line-up to alternate core cooling line-up), the following statement is made: *"Key assumption is that operations explained above are incorporated in the operational procedures, and the related activities are defined as a COL item in DCD Section 19.3 (i.e., COL 19.3(6))."* However, the staff notes that although the referenced COL Action Item states that the COL applicant will develop an "accident management program" nothing is stated about specific operations. Furthermore, it is not clear whether the licensee's "accident management program" will include emergency operating procedures (EOPs) or will address only severe accidentts. Even if EOPs are included, there should be a COL Action Item to ensure that appropriate procedures for important operator actions will be incorporated in the "accident management program." Please explain.

19-360

In response to RAI Question 19-84, the following important design feature is stated:

"Capability exists for a fast depressurization of the RCS (by using the EFW pumps to remove heat through the SGs and by manually opening the MSRVs) to allow alternate core cooling injection using the CS/RHR pumps."

Please include this statement in the list of important "assumptions" regarding design and operational features and state its basis (disposition) by referencing an appropriate DCD chapter or a Chapter 19 thermal-hydraulic analysis.

19-361

Please provide the following information related to your response to RAI Question 19-86 regarding features available in the US-APWR design for preventing interfacing systems LOCA.

- It is stated that "the US-APWR is designed so that the residual heat removal system (RHRS) pressure does not exceed its critical pressure….." However, the basis of this statement is not clearly stated in the response or inferred from the line diagram. Has an analysis being performed to demonstrate that the RHRS pressure does not exceed its critical pressure?

- The presence of relief valves are beneficial but they cannot, by themselves, explain why the critical pressure for pipe break is not reached.
- The line diagram does show paths from the RHRS suction line and the CS/RHR alternate core injection line to the RWSP. Please confirm that there are not any components (not shown in the line diagram) that can restrict the flow. Also, discuss whether the normally open CS/RHR suction isolation valve can be closed due to maintenance and, thus, prevent flow to the RWSP.

19-362

In response to RAI Question 19-87, it is stated that *"…. there are no significant sensitivities on the results of uncertainty analysis even if the EFs used here are different from those reported in NUREG documents."* This statement does not address the staff's question regarding the uncertainty associated with some initiating events, such as loss of CCW (EF = 10), which is a major contributor to the uncertainty of the estimated plant risk. Such areas of uncertainty should be identified in the PRA and taken into account in risk-informed applications.

Also, the assumed partial loss of CCW/ESW frequency may be underestimated, especially if the unavailability of standby pumps due to maintenance is increased compared to operating plants. Were the error factors of initiating event frequencies that are provided In this response used in the updated uncertainty analysis? Please discuss.

19-363

In response to RAI Question 19-88, it is stated that "CCF events are quantified using the MGL method ……. Exceptions are CCF events of components of asymmetric configuration, which involves CCF between standby components and running components." This "exception" results in an arbitrarily chosen value for the beta CCF parameter which is two orders of magnitude lower than what is reported in industry data bases for the CCF of CCW/ESW pumps (set of four pumps). The justification provided for not using industry data is that there are not enough

historical data for the CCF to run of all four pumps in asymmetric configuration (two standby and two running pumps). Since the CCF of all four CCW/ESW trains dominates the total loss of CCW/ESW initiating event frequency, this frequency is an important source of uncertainty for both the base PRA and for risk-informed applications. An investigation of potential root causes of CCF among all four pumps and all four heat exchangers could help narrow this uncertainty since it

is likely that some failures do not impact both CCW/ESW subsystems (they are separated) or both running and standby components. In addition, this source of uncertainty should be identified in the PRA and taken into account in risk-informed applications. Please discuss.

19-364

The responses to RAI Questions 19-97 and 19-98 do not address the staff's question regarding the use in the thermal-hydraulic (T-H) analyses of more than the minimum set of equipment that are required according to the PRA success criteria (e.g., case 1.4 assumes all accumulators, all EFW pumps and all CS pumps are successful when HHI fails). T-H analyses should be run for representative accident sequences using minimum equipment, conservative initial and boundary conditions and limiting assumptions (e.g., about break size and location). For example, the results of a T-H analysis for small LOCA event tree sequence #20 can be used to verify the success criteria and the assumptions regarding the time available to switch from containment spray mode to alternate core cooling mode. Minimum equipment in accordance with the assumed success criteria in the PRA should be used in the analysis.

Please discuss.

19-365

In the response to RAI Question 19-106, it is not clearly stated whether the mechanical failure of the main steam isolation valve (MSIV) and the main feedwater isolation valve (MFIV) to close are modeled in the PRA. Please confirm. Also, please provide the dominant cutsets (e.g., cutsets contributing 99% of the sequence CDF or top 10 cutsets, whichever is smaller) for each of the SGTR sequences #11, #12 and #13, involving steam generator isolation failure.

19-366

In the response to RAI Question 19-108, the following statement is made: *"Operator action to equalize the RCS pressure with the secondary side pressure after isolation of the faulted SG is therefore not modeled …….Operator action to equalize the pressure is assumed to succeed in those sequences."* The staff believes that anytime that the operator is relied upon to perform a task there is a finite probability of failure that needs to be addressed. Please discuss.

In addition, the staff review would benefit from the results of thermal-hydraulic T-H analyses for SGTR sequences #3 and #21 using minimum equipment (in accordance with the assumed success criteria in the PRA), conservative initial and boundary conditions and limiting assumptions (e.g., considering the most severe faulted SG isolation failure mode).

19-367

The following statement is made in the response to RAI Question 19-119: " The PRA considers that by assuming a 0.1 failure probability for MTC, the failure of emergency boration or manual trip is not necessary to be explicitly modeled." However, the staff finds that a simplified modeling of the ATWS accident sequences is used in the PRA without the benefit of any conservative or bounding assumptions. As currently modeled, the risk associated with ATWS sequences is dominated by the following: (1) Support software failure probability of

1E-7; (2) Control rod failure to insert probability of 1E-7; (3) I&C hardware or reactor trip breaker failure to open probability of 3E-6; and (4) a probability of 0.1 for an unfavorable moderator temperature coefficient (MTC) coincident with the success of all trains of several equipment such as all four EFW pumps and all pressurizer SDVs. There is significant modeling and state of knowledge uncertainty associated with the first three probabilities while no basis has been provided for the last probability (it can be significantly higher than 0.1 since one train of EFW can be out for maintenance and one SDV can fail to open). Please discuss. In addition, the ATWS event tree model assumes that the diverse actuation system (DAS) can trip the reactor even when the reactor trip breakers have failed to open. This implies that the implementation of the DAS reactor trip function is through a trip of the control rods via the M-G set field breakers which are separate and diverse from the reactor trip breakers. If the above statement is true, this is an important assumption about a design feature that needs to be verified and the reliability/availability of the M-G set breakers assured (e.g., include in the D-RAP). Please include this statement in the list of important "assumptions" regarding design and operational features and state how it will be verified by referencing an appropriate DCD chapter and/or associated requirements.

19-368

In the response to RAI Question 19-122, the features and characteristics of the RWSP strainers that prevent debris from plugging valves are discussed. This information should be included in the list of important "assumptions" associated with design and operational features. Also, in the response to RAI Question 19-123, the "assumption" in the PRA of diverse configuration of check valves in the various injection lines is discussed. This information should be included in the list of important "assumptions" associated with design and operational features since the assumed diverse configuration minimizes intersystem CCF and, therefore, such CCFs were not modeled in the PRA. Please discuss.

19-369

In the response to RAI Questions 19-126 and 19-128, the issue of consistency between the assumed demand failure rates and the US-APWR testing intervals is discussed. It is stated that "….testing interval was not used in calculation of reliability of equipment for which NUREG/CR-6928 data was applied." The staff believes that the testing interval of standby equipment is a major factor in estimating failure probabilities of equipment modeled in the PRA. The NUREG/CR-6928 failure data, used in the US-APWR PRA, are based on operating reactor testing intervals, which in many cases are significantly different than the testing intervals proposed for US-APWR. Please discuss an approach for addressing this issue.

Also, it is stated that a list of test intervals will be added for each system in the next revision of the PRA technical report. However, it is also stated in the response that only "test intervals that have technical basis will be added for each system in the next revision of PRA technical report." Please clarify.

19-370

In the response to RAI Question 19-126 it is stated that "…valve position indication is credited for motor-operated valves that need to maintain their position during standby." In the response to RAI Question 19-132 it is stated that "Valve positions will be checked from the main control room after outages or testing. Valves that have been aligned in the wrong position will be detected and fixed to the correct position within a short period of time. The probability of valves to be left in the incorrect position after testing is therefore considered to be very low. Accordingly, human error of omission …. is not modeled in the PRA." Since the risk contribution of human errors during testing and maintenance can be significant if not detected and corrected, all assumptions about early detection of failures made in the PRA should be stated for each system. In addition, the response to RAI Question 19-132 should be included in the list of PRA "assumptions" regarding important design and operational features and be properly verified for the as-to-be-built, as-to-be-operated plant.

19-371

In the response to RAI Question 19-133 it is stated that "Equipment of new built plants are considered to have enhanced reliabilities compared to existing plants. " However, more evidence is needed to support this statement. The failure rates reported in NUREG/CR-6928 are average values that are mostly based on components tested quarterly. The two pairs of motor-operated valves of the charging system, discussed in the response to RAI Question 19-133, are tested every 24 months at refueling. The performed sensitivity analysis does not address the uncertainty associated with the applicability of NUREG/CR-6928 data to MOVs with much longer testing intervals or the fact that pre-1997 failure data were not included in NUREG/CR-6928. The following two sensitivity cases could be helpful to determine whether the NUREG/CR-6928 data need to be adjusted for the longer testing intervals of the US-APWR:

Case 1: Use NUREG/CR-6928 data after adjusting them for the 24-month testing interval.
Case 2: Use data from EPRI's "ALWR Utility Requirements Document," which are averages of data taken from several sources.

Similar sensitivity studies are needed for EFW system components tested every 24 months (RAI Question 19-183). Please discuss.

19-372

In the response to RAI Question 19-186 it is stated that "In addition, if the plant were to continue hot standby condition, the two emergency feedwater (EFW) pits together contain enough water volume to maintain hot standby condition for 24 hours without water supply." However, if an operator action is required at some time after 24 hours to maintain the stable plant condition (e.g., operator action to supply water from the demineralized water storage tank), this action should be modeled in the PRA, unless it is shown to be insignificant. Please discuss.

19-373

In the response to RAI Question 19-187 it is stated: *"Even if the manual valves were to be tested with long intervals, the increase in failure probabilities of manual valves due to long test intervals is considered to be much smaller than the human error probability to operating valves. For this reason, the use of the generic data was considered not to have impact on system reliability."* Please explain how the failure probability of manual valves with no testing requirements is "much smaller" than the human error probability of 4E-3. According to event EFWOO01PW2AB (operator failure to change over to the other EFW pit), there are three manual valves (PW2A, PW2B, and PW3) that can fail to open and two manual valves (PW1A and PW1B) that can fail to close. Please explain how these failures were modeled.

19-374

In the response to RAI Question 19-188 it is stated: "For this reason, failure to control the valves during standby are not modeled. CCF of the control valves to control during the mission time needs to be considered and such failures will be incorporated in the PRA during the next PRA upgrade." Please explain the reason that the EFW line throttle MOV failure to "control" needs to be modeled during the 24-hour mission time but not during standby. If these valves are not needed to control the flow to the SGs during an accident, as it is stated in the response, then why does their failure to control flow during the 24-hour mission of the EFW system need to be modeled in the PRA?

In addition, in the response to RAI Question 19-190 it is stated that "…water level in the SGs needed for secondary cooling can be maintained by the actuation of interlocks implemented on the EFW control valves … and the EFW isolation valves….." This statement should be included in the list of PRA "assumptions" regarding important design and operational features and be properly verified for the as-to-be-built, as-to-be-operated plant.

19-375

In the response to RAI Question 19-189 it is stated that no CCF of EFW pit water level sensors are modeled because (1) water level is checked every 12 hours, (2) failure of a sensor can be detected and repaired by recognizing any inconsistent output signals between the two sensors, and (3) CCF of both sensors is unlikely. Please explain by responding to the following questions:

- Is there a requirement that the operators must check the EFW pit water level every 12 hours?
- How likely is it that the reading from a failed sensor will not be significantly different than the reading from the not failed sensor in a 12-hour period?
- Why is the failure of a sensor during the 24-hour mission time modeled in the PRA but
  pre-existing failures during the 12 hour period between checks are ignored?
- Could the failure of both sensors result in the same or close readings and go undetected? What is the sensor failure mechanism?

It is stated that CCF during the 24-hour mission time will be incorporated even though it does not impact the PRA results. The staff believes that some failures that do not impact the base PRA results may impact the results of risk-informed applications (e.g., risk-managed TS). For this reason, a good quality PRA includes even failures that do not impact the base PRA results.

It is stated that a miscalibration error across all sensors (i.e., sensors in both EFW pits) is not considered because the two pits are located in opposite sides of the reactor building and, therefore, the operator action failures have very low dependency. However, it is more likely that the operator calibration error is due to incorrectly following the same procedure. Please discuss.

19-376

In the response to RAI Question 19-191, the following statements are made:

(1) It is more likely that the actual out of service time will be determined by failure type and repair time with an expectation that the newer design pumps will experience higher reliability.

(2) The actual outage times are also expected to be impacted by regulations such as MSPI and derivative requirements that impact availability monitoring.

(3) A sensitivity study, assuming a seven day yearly outage of each EFW pump, indicates an increase in CDF of about 9%.

The staff views this issue as a source of uncertainty that needs to be tracked and taken into account in risk-informed applications involving decision-making. Please discuss.

19-377

In RAI Question 19-195 the staff requested that a systematic search be performed to identify missing failures in all fault trees used in the PRA. The response to RAI Question 19-195 addresses only the examples of missing failures provided by the staff. Please perform a systematic search to identify failures that are not modeled in the PRA and either incorporate these failures in the revised fault trees or explain in the fault tree assumptions the reasons these failures are not modeled.

Also, it is stated in the response to RAI Question 19-195 that no credit is taken for manual actuation when automatic actuation fails because the probability of I&C equipment failure is negligible. However, the assumption of “negligible” failure probability of I&C hardware and software may not be robust. Furthermore, a PRA which is a living document that will be used for sensitivity analyses and risk-informed applications, should model even those failures that do not appear to be risk significant in the base PRA. Please discuss.

In addition, it is stated in the response to RAI Question 19-195 that the “failure to open” of the motor-operated main steam relief valve (MSRV), to depressurize the secondary side, is not modeled because it is expected to be much lower than the human error to open the valve which is 2E-3/d. However, the failure probability to open a motor-operated valve is assumed to be 1E-3/d in the US-APWR PRA and higher in other sources. Please discuss.

19-378

In RAI Question 19-196 the staff requested that the assumptions made in the PRA about test and maintenance be listed for all components modeled in the PRA. However, the response addresses only the examples provided by the staff. Please include the requested information in the next revision of the PRA.

19-379

In the response to RAI Question 19-197 it is stated that the issues brought up by the staff regarding the failure of the turbine bypass valves (TBVs) will be addressed. However, it is not stated in the response how and when these issues will be addressed. Please clarify.

19-380

The following statement is made in the response to RAI Question 19-198: "The PRA model will be upgraded to properly assess the [a]symmetrical condition of initiating events when applying to RMTS." However, it is not stated when this upgrade will take place and whether it will be performed at the DC stage or later. The issue of simplifying modeling assumptions and their impact on PRA results and insights, which are used in risk-informed applications for decision making, should be one item in the list of issues to be addressed before the US-APWR PRA can be used to support applications such as the risk-managed technical specifications (RMTS) program. Please clarify.

19-381

Please provide the following information related to your response to RAI Question 19-200:

(a) It is stated: *"Isolation of ruptured SG can be accomplished without operator action. Operator action is [needed] to isolate the ruptured SG [and] is credited when the TBVs fail to reclose."* This statement appears to conflict with statements made elsewhere in the PRA (e.g., Section 3.2.5.1 of PRA Rev.1 "Event Description" states that "MSIV will be manually closed" without any reference to the TBVs. Also, on page 3-25 it is stated that "Operator action to close MSIV" is needed, independently of what happens to the TBVs). Please clarify.

(b) In explaining the frequency of SGTR sequence #12, the potential failure to re-close of only one TBV is considered (event MSRAVCD500A1 with probability of 1.2E-3). However, there are 15 TBVs and each one of them can fail to re-close and isolate the ruptured SG. Please discuss.

(c) A low dependency is assumed between the operator action to recognize the need and close the MSIV associated with the ruptured SG (event MSPOO02533A) and the operator action to recognize the need and close the manual valve that isolates a TBV that failed to re-close (event MSPOO0250A1-DP2). One of the reasons for this low dependency is the assumption that two different crews will be performing the two actions. However, it appears that the same crew is performing the cognition aspects for both actions. In addition, it is not clear what the definition of a second crew is in the US-APWR PRA. Please discuss.

(d) It is stated that if there is a third human error in a sequence, then its dependency is at least moderate. However, in the US-APWR PRA this dependency is always assumed as moderate. Please discuss.

(e) The failure of the main steam safety valves (MSSVs) to reclose due to passage of water was not modeled in the PRA. It is argued that the probability of an MSSV to stick open after passing water is 0.1 according to NUREG/CR-6928. However, other sources put this probability close to 1. Also, it is stated in the response that an MSSV will open only when the air-operated (main steam relief valve) MSRV fails to open on demand, which contradicts the PRA modeling assumption that two MSSVs open when either the turbine bypass valves (TBVs) fail to open or the MSRV fail to open soon after the SGTR event. Please discuss.

19-382

In the response to RAI Question 19-272(b) it is stated that the generic data for gas turbine generators (GTGs) in NUREG/CR-6928 are based on only two components at the same plant that are not safety-related. For this reason, it is stated that these data do not apply to the safety-related emergency GTGs used in the US-APWR design. This is a reasonable argument. However, this argument raises a question regarding the failure rates used for the alternate ac (AAC) GTGs which also are not safety-related. Please discuss.

In addition, the staff's question regarding the performed sensitivity studies was partially answered. A response to the following question is still expected: Provide the basis for the assumption that the CCF parameters of GTGs are not likely to be higher than the CCF parameters of "general components," used in the sensitivity analysis.

19-383

Please provide the following information related to your response to RAI Question 19-275:

(a) It is stated that the turbine-driven (T-D) emergency feedwater (EFW) pump "is designed to operate without HVAC for several hours" and that "…recovery of core cooling by RHR is expected during this time." The term used for the length of time a T-D pump can operate without HVAC (i.e., "several hours") is ambiguous and can take several interpretations. How does “several hours” compare to the mission time of 24 hours used in the PRA? Is “several hours” applicable to all accident sequences that are considered a “success” in the PRA? Please discuss.

(b) It is stated that the loss of HVAC in the main control room (MCR) is not modeled because operator actions can also be performed with the remote shutdown console (RSC). This argument would be reasonable if it was supported by qualitative arguments, such as regarding the probability of MCR HVAC failure, the failure probability to transfer control to the RSC, and the operator actions modeled in the PRA which cannot be performed from the RSC (if any). Please discuss.

19-384

In the response to RAI Question 19-276, regarding the staff’s question about the assumed Failure rate of 7E-5 per demand for a stuck open pressurizer safety valve (PSV), it is argued that the PSVs were conservatively assumed to always open following an initiating event. However, the staff believes that this kind of “compensation” should not be used in the PRA, especially if the PRA is going to be used for sensitivity studies and risk-informed applications. Please discuss.

19-385

Please clarify and provide the basis for the following statement made in the response to RAI Question 19-279: *“We understand that none of the events used to calculate the MGL parameters in NUREG/CR-5497 were applicable to normally operating pumps and none of the events, if assumed to occur in a system of normally operating pumps, would have any significant potential for leading to failure of any, even a single normally operating pump.”*

19-386

In the response to RAI Question 19-280, the basis for the assumed time windows used in the human reliability analysis (HRA) is provided. These time windows are primarily based on thermal-hydraulic (T-H) calculations reported in Appendix 5A of the PRA report. However, as stated in other follow-up RAIs related to PRA success criteria, more than the minimum set of equipment that are required according to the PRA success criteria is used in the calculations. For example, the results of T-H analyses for medium LOCA sequence # 29 and small LOCA sequence #20 can be used to verify the assumption regarding the time available to switch from containment spray mode to alternate core cooling mode. Minimum equipment in accordance with the assumed success criteria in the PRA should be used in the analysis. Please discuss.

19-387

Please provide the following information related to your response to RAI Question 19-283:

(Answer to a): Does the following statement reflect a PRA assumption? "The availability and reliability of all trains of safety-related systems will be controlled by the maintenance and configuration risk management programs. Availability goals will be set for each train of all safety-related systems and their availability will be tracked and compared to these goals." Please clarify.

(Answer to c): It is stated that "Risk important operator actions will be included in table 19.1-115…" The list of risk-important operator actions does not need to be included in Table 19.1-115 but be identified as a COL Action Item in Chapter 19 to provide this information in other DCD chapters (e.g., DCD Chapter 18 discusses the use of this information in developing and implementing procedures, training and other human reliability related programs for the plant). The staff requested a systematic search to identify "assumptions," in terms of design and operational features and associated requirements, made in the HRA and, for each of these assumptions, indicate how it will be ensured that they will remain valid for the as-to-be-built, as-to-be-operated plant.

Answer to (d): Sensitivity Case 3-3 does not provide any useful insight because an arbitrary failure probability was used for each of the sump screens and independence was assumed. Information from the resolution of GSI-146 (sump clogging in PWRs) should be used to identify bounding parameters to be used in a meaningful sensitivity study, if possible, or to identify requirements for the design or for the operation of the plant that would minimize the risk from sump clogging. Please discuss.

Answer to (e): How is the non-safety digital I&C system software "independent" from the application software used for the safety systems? Is it diverse from the software used in the safety digital I&C? Is the reliability of the non-safety digital I&C software comparable to that of the safety digital I&C software? Please discuss.

19-388

Please provide the following information related to your response to RAI Question 19-285:

Answer to (a): It is stated that a sensitivity study was performed where the failure probabilities of the accumulator injection check valves were increased by an order of magnitude and the result was an increase in CDF by about 3%. The staff concern is for accident sequences where accumulator injection is credited while the primary system pressure may be still relatively high, such as SLOCA sequence # 29, VSLOCA sequence # 21, PLCCW sequence #31 and LOOP sequence #33 (in Sheet 1). For such cases, an order of magnitude increase (from about 1E-5 to about 1E-4) of the check valve CCF probability in the sensitivity study may not be adequate to address the uncertainty. Please discuss.

Answer to (b): The last sentence of the response reads as follows: *"If the probability of software common cause failure that results in failure of all safety related signals modeled in the PRA is assumed as higher than the probability of the application software common cause failure of the base case, the CDF results in approximately 1.5 time higher than the base case."* Please clarify.

19-389

Please provide the following information related to your response to RAI Question 19-290:

Answer to (a): The statement *"non-safety related SSCs are designed or located to avoid adverse interaction with safety-related SSCs"* should be included in Section 19.1.7.1 (Table 19.1-65) of the DCD with reference to the related analysis provided in DCD Section 3.2.

Answer to (b): Identify those relay features and characteristics which ensure that relay chatter does not occur or refer to analyses in other DCD chapters which show that safety functions are not affected even if chatter does occur.

Answer to (e): Explain why a fragility analysis is not needed at the design certification stage. How can we be certain that an SSC that is designed for 0.3g pga SSE will have a HCLPF value of at least 0.5g?

Answer to (f): It is stated that the probability that all gas-turbine generators (GTGs) fail to run for 24 hours is 9.9E-4 and, therefore, it is below the cutoff value of 1E-3 for mixed cutsets. However, this probability is 1.15E-3/yr (1.6E-4 for first hour and 9.9E-4 for remaining 23 hours). In addition, the mission time in this case may be longer than 24 hours. Please discuss.

Answer to (g): The min-max method assumes that if an earthquake causes the failure of an SSC of a certain HCLPF value, this earthquake also causes the failure of any other SSC with equal or less HCLPF value. Therefore, the combination "seismically induced SLOCA AND seismically induced failure of the T-D EFW pumps AND random failure of the M-D EFW pumps" is not a mixed cutset because the combination of the two seismically induced failures (i.e., the first two terms) alone is a cutset. Please discuss.