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Framework for AP600
Severe Accident Management Guidance

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Enclosure 1

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TABLE OF CONTENTS

1.0 INTRODUCTION

Prevention and mitigation of accidents has been an integral part of the design process for AP600. A significant driving force in the passive plant design is the key accident management philosophy of preventing accidents from progressing to core damage. However, in the event of a low probability core damage accident, it is prudent to have severe accident management guidance with the objective of terminating the progression of the accident and returning the plant to a controlled, stable state. Therefore, this document contains a summary of the overall philosophy and high level strategies that will form the basis of the AP600 severe accident management guidance.

The Westinghouse plan for addressing severe accident management for AP600 will be based on the Westinghouse Owners Group Severe Accident Management Guidance (WOG SAMG) for the current generation of operating plants. Since some of the AP600 design features reduce or eliminate the potential for some severe accident phenomena and fission product boundary challenges, the WOG SAMG provides an envelope of possible severe accident management considerations. Thus, the WOG SAMG has direct applications to the development of AP600 severe accident management guidance, and will be the starting point from which comparisons are made.

The scope of the AP600 severe accident management guidance is to address significant core damage accidents. Prior to core damage, the Emergency Operating Procedures (EOPs), which are based on the AP600 Emergency Response Guidelines (ERGs) will be used. [Ref. 1] Although the EOPs/ERGs for existing plants have proven to be effective in the prevention of core damage, they do not address scenarios after significant core damage has occurred.

The AP600 severe accident management guidance will be developed for use after the AP600 emergency response guidelines are no longer applicable. The AP600 severe accident management guidance will include the application of insights that are derived from the AP600 Probabilistic Risk Assessment (PRA), and elements that have been learned through severe accident management research over the past 15 years. As such, severe accident management guidance is the mechanism that brings the current level of knowledge on severe accidents to the hands of the operating and technical staff at the plant. However, the overall uncertainty of the core melt progression is still quite high, and thus the management of a severe accident can only be pre-constructed by guidance that is less prescriptive than the guidelines for design basis events and other accidents prior to core damage.

The contents of this document include a discussion of severe accident management requirements, the anticipated structure for the decision making process, the goals that must be accomplished for severe accident management, and a summary of possible strategies for AP600 severe accident management. This document provides the framework for future AP600 severe accident management guidance development and therefore does not specifically address many issues in detail.

2.0 REQUIREMENTS FOR SEVERE ACCIDENT MANAGEMENT

There are no current NRC requirements for the development of severe accident management guidance. However, NRC policy statements from the NRC Staff to the NRC Commissioners (SECY letters) identify concerns and future actions of the NRC concerning this subject. Specifically, SECY-89-012 provides the following information.

> "Accident Management encompasses those actions taken during the course of an accident by the plant operating and technical staff to: (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) maintain containment integrity as long as possible, and (4) minimize offsite releases. Accident management, in effect, extends the defense-in-depth principle to plant operating staff by extending the operating procedures well beyond the plant design basis into severe fuel damage regimes, with the operator skills and creativity to find ways to terminate accidents beyond the design basis or to limit offsite releases.
>
> The NRC staff has concluded, based on PRAs and severe accident analyses, that the risk associated with severe core damage accidents can be further reduced through effective accident management. In this context, effective accident management would ensure that optimal and maximum safety benefits are derived from available, existing systems and plant operating staff through pre-planned strategies... Accordingly, accident management is considered to be an essential element of the severe accident closure process described in the Integration Plan for Closure of Severe Accident Issues (SECY-88-147) and the Generic Letter on the Individual Plant Examination (Generic Letter 88-20).
>
> In the IPE Generic Letter, the staff deferred the requirement to develop an accident management plan, stating that we are currently developing more specific guidance on this matter and are working with NUMARC to (1) define the scope and content of acceptable accident management programs, and (2) identify a plan of action that will ultimately result in incorporating any plant-specific actions deemed necessary, as a result of the IPE, into an overall severe accident management program."

Also within SECY-89-012, the first objective for an accident management plan developed by licensees for each plant is:

> "Developing technically sound strategies for maximizing the effectiveness of personnel and equipment in preventing and mitigating potential severe accidents. This includes ensuring that guidance and procedures to implement these strategies are in place at all plants."

In a June 3, 1993 meeting between the NRC and NUMARC, the NRC endorsed a NUMARC industry initiative on accident management as an alternative to an NRC Generic Letter. [Ref. 2] Currently in draft form, the industry initiative does not contain specific regulatory criteria. Rather, industry has defined its goals and objectives by its actions relative to severe accident management. These include, but are not limited to, performance and submittal of an Individual Plant Examination (IPE) and an Individual Plant Examination of External Events (IPEEE), development of generic (Owners Group) severe accident management material, and numerous interactions at various levels among industry, the NRC, and vendors. It is anticipated that the NUMARC initiative will be issued in March 1994 as a revision to NUMARC 91-04, "Severe Accident Issue Closure Guidelines." The NRC has expressed the view that the industry initiative is acceptable as an alternative to the accident management generic letter provided that the NRC could conclude that "the initiative meets the objectives for accident management established at the time the severe accident program was initiated." [Ref. 2]

The previous information is in regards to general positions of the NRC on severe accident management, and it does not distinguish between current operating plants and new, advanced plant designs. However, the NRC has indicated their interest in AP600 severe accident management through several of the Requests for Additional Information (RAIs) over the past year. Specifically, RAI 720.55 asks how Westinghouse plans to use the AP600 Probabilistic Risk Assessment to identify and assess accident management measures. Furthermore, in RAI 720.56, the NRC asked how Westinghouse plans to address the five elements of accident management as defined in SECY-89-012. These elements are: 1) accident management procedures, 2) training for severe accidents, 3) accident management guidance, 4) instrumentation, and 5) decision-making responsibilities.

In addition, the Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD) states that the Plant Designer shall establish the technical basis for a severe accident management program that includes core damage prevention and mitigation. The Plant Designer is also to translate the plant design bases into operational limitations and responses which can then be developed into procedural guidelines and training by the Plant Owner. The Plant Designer is also responsible for confirming that the plant design is compatible with the EPGs and the severe accident management program based on the plant specific PRA and other relevant information. [Ref. 3] The NRC's Safety Evaluation Report for this document states: "The use of PRA for developing and confirming the severe-accident management program and EPGs is also consistent with the Commission's severe-accident policy." [Ref. 4]

3.0 DECISION-MAKING PROCESS

Severe accident management involves the implementation of actions to bring the plant to a controlled, stable state following core damage and to mitigate challenges to the containment fission product boundary. In a severe accident state, the first two fission product boundaries (the fuel rod cladding and the reactor coolant system) may be severely damaged and the focus shifts to maintaining the final fission product boundary. To effectively choose the appropriate severe accident management actions and to prioritize the implementation of the appropriate actions, assessment of the plant conditions is needed.

The nature of severe accidents and the possible responses dictate that severe accident management diagnostics be symptom-based. Several specific features of severe accidents can be cited which support the symptom-based approach:

a) Severe accident management must provide a response for a wide range of severe accident conditions. While a large number of possible scenarios have been identified in severe accident studies, it is likely that most of these scenarios do not accurately represent realistic severe accident scenarios due to modelling assumptions in these studies (such as all equipment failures are assumed to occur at time zero).

b) During a severe accident, the plant conditions are undergoing continual change. Severe accident management must relate actions to symptoms.

c) The overall goals of severe accident management involve the response to challenges to fission product boundaries, which can be diagnosed through symptoms.

In other words, the symptom-based approach is a key method to develop flexibility in the AP600 severe accident management guidance. This flexibility refers primarily to the ability of plant personnel to shift priorities and implement accident management strategies based on the situation of the plant during the accident. Specific technical decisions may be knowledge-based, and will be dependent on the interpretation of the plant status. Therefore, the appropriateness of specific actions cannot be predetermined during the development of AP600 severe accident management guidance. This approach allows the guidance developed to be useful during any severe accident, even scenarios which are not currently recognized situations. As such, an AP600 severe accident management plan is the final stage in the defense-in-depth plant safety concept.

Although flexibility is a necessity, there is a need for the guidance itself to be a <u>structured</u> process for choosing the appropriate actions based on actual plant conditions. Human factor considerations during a high stress environment that would accompany a severe accident require that the guidance be simple to use. Thus, the AP600 severe accident management guidance must be an effective decision-making tool based on some fundamental concepts about the organization of the guidance, as detailed below.

3.1 Role of the Plant Personnel

NUMARC has developed recommendations for severe accident assessment and mitigation that divide responsibilities of personnel into categories of *evaluators, decision makers,* and *implementers.* The evaluators must assess the plant symptoms to determine the plant state, and then evaluate the potential strategies that may be used to mitigate the event. The decision makers are to assess and select the strategies to be implemented. The implementers are responsible for performing the steps necessary to accomplish the objectives of the strategies, such as hands-on control of valves, breakers, controllers and special equipment.

The plant personnel to perform each of these functions will be identified in a later phase of the development of the AP600 severe accident management plan. Factors that will be considered include:

- the structure of the organization that is needed for accidents prior to core damage, so that there would be an orderly transition to management of the accident after core damage is diagnosed,
- the instrumentation, equipment and computers necessary to fulfill each function,
- the skills, training and expertise of personnel,
- the size and location of the necessary staff, and
- the desire to address severe accident management preparation, while still maintaining a focus on the prevention of core damage.

3.2 Structure of AP600 Guidance

The AP600 guidance for severe accident management will include overall diagnostic tools that control the flow of the decision-making process, as well as detailed guidelines. The following sections provide a summary of the expected flow charts, as well as further information on the content of the detailed guidelines.

3.2.1 Diagnostic Flow Chart

As identified in Section 3.0, there is a need for severe accident management guidance to have an organized structure to facilitate effective decision-making. For AP600, the form of this structure should be based on the WOG SAMG, although some of the details may differ. The element discussed within this section is the Diagnostic Flow Chart, which is the primary decision-making tool to determine when the plant has achieved the overall goals of severe accident management.

The Diagnostic Flow Chart (DFC) is the primary tool to identify the appropriate guidelines for the key possible plant conditions that may occur following a severe accident. The flow chart is the point of entry into severe accident management (from the ERGs), and it also serves as the exit point. The flowchart is based on setpoints for different parameters that are either necessary to define a controlled, stable state or which may prevent further challenges to fission product boundaries. The elements that determine a controlled, stable state are discussed in Section 4.0. Prevention of fission product boundary challenges refers to the prevention of severe accident phenomena, which may challenge fission product boundary integrity, such as induced steam generator tube rupture, high pressure melt ejection and reactor vessel lower head failure. Key plant conditions will be defined based on the capability to take actions to control the conditions and on the potential challenge to the containment fission product boundaries which these conditions may indicate. Based on the particular plant conditions identified in the DFC, a specific guideline is consulted to evaluate the availability and effectiveness of the various severe accident management strategies which may be used to control the conditions. If a controlled, stable state is achieved, the DFC instructs plant personnel to develop a set of limitations and cautions for the long term recovery process, based on the consideration of large quantities of fission products released from the core and other important aspects of the severe accident scenario. The parameters in the DFC will be prioritized and the setpoint values will be determined during the development of the detailed AP600 guidance.

3.2.2 Severe Challenge Status Tree

The Severe Challenge Status Tree (SCST) is the primary tool used by the emergency response team to identify immediate and severe challenges to containment fission product boundaries and to select the appropriate guideline for strategies to respond to the challenge. The SCST identifies the severe challenges for all possible plant conditions that may occur following a severe accident. The plant conditions on the SCST will be defined based on the severity of the challenge and capability to take actions to control the conditions in time to mitigate the challenge to the containment fission product boundaries. Based on the particular plant conditions identified in the SCST, a specific guideline is consulted to evaluate the availability and effectiveness of the various severe accident management strategies which may be used to control the conditions.

The parameters in the SCST are regularly monitored to determine whether a severe challenge has developed. The SCST parameters are to be monitored simultaneously with the usage of the Diagnostic Flow Chart (DFC). The existence of the SCST as a monitoring tool allows for the effective use of the pre-prioritized DFC, which addresses less-immediate concerns. However, if the setpoint for a SCST parameter is reached, all activities being guided by the DFC would be put on hold until the SCST challenge has been addressed.

3.2.3 Guidelines

While a Diagnostic Flow Chart and Severe Challenge Status Tree are used to establish the organizational structure of severe accident management guidance, the details and the majority of the technical content are contained within guidelines. Guidelines are referenced directly from the DFC or SCST due to a plant parameter being outside the desired range. The structure of the guidelines will include the following major considerations:

1) **Equipment Availability** - The guidelines will contain lists of the possible equipment that may be used to implement an action. If no equipment is available, instructions will include the consideration of restoring the non-functioning equipment.

2) **Benefits vs. Potential Negative Impacts** - The potential actions will be considered in regards to their benefits weighed against the expected negative impacts. If the negative impacts are judged to be large, then methods to minimize the negative impacts will be considered when possible. If the impacts differ based on the choice of methods or equipment, this distinction will be made.

3) **Implementation** - If the decision is made to implement a strategy, implementation instructions will be provided that include any limitations that were identified during the evaluation. The implementation instructions will also identify the expected response of the plant as a basis to compare the actual response. The option to abort the action, or to implement additional actions, will also be considered.

4.0 SEVERE ACCIDENT MANAGEMENT GOALS

Before any guidance for severe accident management can be developed, the first step is to identify the overall goals that the guidance must achieve. As stated in the introduction, the overall objective of severe accident management is to terminate the core damage progression. However, the scope of severe accident management also entails maintaining the capability of the containment as long as possible, and minimizing fission product releases and their effects. These severe accident management objectives can be translated into specific goals that must be met. These three goals are: 1) to return the core to a controlled, stable state, 2) to maintain or return the containment to a controlled, stable state, and 3) to terminate any fission product releases from the plant. Secondary goals, to be achieved while focusing on the primary goals, are to i) minimize fission product releases, and ii) maximize equipment and monitoring capabilities.

Before details are provided on each of these goals, it should be noted that severe accident management does not guarantee the achievement of the goals. Severe accident management is a structured approach that best utilizes available resources at the plant based on the current understanding of severe accidents.

4.1 <u>Controlled, Stable Core State</u>

A controlled, stable core state is defined as core conditions under which no significant short term or long term physical or chemical changes (i.e., severe accident phenomena) would be expected to occur. A significant short term or long term change is one which would require an operator response to prevent a change in core location, a challenge to containment integrity, or fission product releases. In order to achieve a controlled, stable core state, two primary conditions must be met.

1. A process must be in place for transferring all energy being generated in the core to a long term heat sink.

2. The core temperature must be well below the point where chemical or physical changes might occur.

For a severe accident, the core is assumed to be uncovered and overheated when severe accident management begins. Therefore, both decay heat and sensible heat must be removed from the core, along with any chemical heat which is produced during the recovery phase. However, providing a means to remove all of the core energy does not guarantee a controlled, stable core state. This is best illustrated by the TMI-2 accident in which core relocation continued for a significant period of time after a process was in place for cooling the core [Ref. 5]. This was because the core geometry did not facilitate efficient transfer of energy from the molten core material to the coolant. Thus, the core can only be considered in a controlled, stable state when its temperature is sufficiently low, <u>and</u> a heat removal process is in place. Thus, both criteria are necessary and sufficient conditions for achieving a controlled, stable core state.

The amount of energy that will have to be transferred from the core is dependent on whether the core remains subcritical. Before significant downward relocation of core material occurs, the amount of negative reactivity required for subcriticality is bounded by the ERG considerations. As core downward relocation progresses, the required negative reactivity for subcriticality decreases due to geometry compaction [Ref. 6]. The core compaction results in a significant change in the local moderator to fuel volume ratio, thus requiring less negative reactivity such as control rods or soluble boron. a,c

However, for severe accident management, the extent of core relocation cannot be determined during the accident itself. If the water injected during the severe accident comes solely from the tanks inside the containment that are sufficiently borated, then there is no chance that the shutdown margin will be lost. However, if the only available water sources do not contain sufficient boron to ensure that the subcriticality conditions are achieved, there is the potential for a return to power, depending on the core geometry. The use of unborated (or under-borated) water could only result in a return to power in the core at very low levels, which is a function of the injection rate to the core. For this scenario, the core would be likely to continue to degrade since all of the heat generation is not removed by boiling of the injected water, resulting in a change in core geometry which leads to a subcritical state.

If the core returns to a critical state, the excess reactivity would be compensated by void formation in the water. However, the rate at which criticality is approached must be sufficiently slow that the feedback associated with the void development can be effective. If the injection rate of the water were too high, prompt recriticality could be a concern, which could damage reactor coolant piping or steam generator tubes. However, generic severe accident studies [Ref. 6] have conservatively shown that even flow rates of 1000 gpm are an order of magnitude too low for prompt criticality. Since this is higher than any expected injection flowrates for the AP600 plant, there is no need to further consider criticality or prompt criticality issues.

The cooling of the core can be accomplished via several methods. The preferred method is to cover the core debris with water while it is still in the reactor vessel. If the core cannot remain covered with water while in the vessel, submerging the bottom head of the reactor vessel with water may be sufficient to remove the core heat. [Ref. 7 and Ref. 8] If this method of flooding the containment cavity is successful, it prevents RPV failure and movement of the core material into the containment. Although either water inside the RPV or water submerging the bottom head of the RPV may be sufficient, the ideal condition is to create water inventories both inside and outside the RPV. This maximizes the possibility of reducing the core temperature and ensuring that further physical and chemical changes can no longer occur.

If the core remains within the RPV, not only must the core initially be cooled, but a long term heat removal process must be established. The first possibility to be considered is heat transfer to the steam generators. For this option to be feasible, there must be a water inventory in the secondary side of the steam generators, the RCS should be relatively intact, and there must be some water inventory within the RCS. However, it is not necessary to have a complete RCS water inventory, since condensation of steam is also an effective heat transfer mechanism.

Another possibility for long term heat removal while the core is within the reactor vessel is to use the passive residual heat removal (PRHR) system. This system is based on natural circulation from the RCS to heat exchangers in the in-containment refueling water storage tank (IRWST). Within the IRWST, the heat is then transferred to the containment through steaming. Therefore, the PRHR is an indirect method to transfer the core heat to the containment. For the PRHR to function, the RCS must be relatively intact, and there must be some water inventory within the RCS. In addition, there must be a sufficient water inventory within the IRWST. Since the IRWST is the largest water source for refilling the RCS and to flood the containment cavity, the IRWST water inventory is not likely to be maintained during a severe accident, and thus this method is not likely to be available for long term heat removal from the RCS.

The third long term heat transfer process to be considered is a direct path to the containment, which is then cooled through passive containment cooling. If there is a LOCA, steaming from the break can be an effective heat transfer medium, provided that additional water can continually be provided to the RCS. For a non-LOCA transient, an opening in the RCS can be created for direct steaming, such as opening the fourth stage valves of the automatic depressurization system (ADS). Another heat transfer pathway to the containment is via direct heat transfer through the walls of the RPV, coolant loops and direct vessel injection lines if water is surrounding the outside surfaces.

If the severe accident is not mitigated before the RPV lower head fails and the core debris is transported ex-vessel, the only long term heat sink is the containment. In this scenario, a water inventory in the containment is needed for initial core cooling and long term heat removal. If the limited surface area of the core debris is not sufficient to permit removal of decay heat and sensible heat, the core debris will remain molten and lateral movement will increase the heat transfer area until cooling can occur. One of the features of the AP600 plant is that the reactor cavity has been designed with sufficient floor space to permit debris spreading until a coolable geometry can be created. Thus, the cooling of the core debris external to the reactor vessel can be accomplished with the presence of water. In addition, accident sequences with debris quenching and a continual supply of water to the cavity experience very little, or no, concrete ablation.

For core material dispersed at reactor vessel failure and refrozen on vertical containment surfaces and equipment, or present as thin layers on horizontal containment surfaces or equipment, no water may be required for long term cooling. Generic analyses show that convection of the decay heat to the containment atmosphere could be sufficient to ensure long term cooling. If decay heat cannot be removed by convection, the dispersed core material will heat-up, become molten, and eventually drain to lower levels of the containment. Downward relocation of core debris will stop when all of the heat can be removed, either via convection from a new configuration or via transfer to water if the debris becomes submerged at lower containment levels. Furthermore, AP600 is designed such that only a small fraction of the core debris that is ejected from the reactor vessel could reach the upper containment area. [Ref. 7]

Therefore, core coolability after vessel failure remains primarily a concern of establishing a water inventory in the lower cavity.

Thus, to maximize the possibility of achieving a controlled, stable core condition, the elements that must be considered in severe accident management are:

- water inventory in the RCS,
- water inventory in the containment cavity,
- heat transfer to the steam generators, and
- heat transfer to the containment.

4.2 Controlled, Stable Containment State

A controlled, stable containment state is defined as containment conditions under which no significant short term or long term physical or chemical changes would be expected to occur. A significant short term or long term change is one which would require an operator response to prevent a challenge to containment integrity or fission product releases. In order to achieve a controlled, stable containment state, several conditions must be met, as summarized below.

1. A process must be in-place for transferring all of the energy that is being released to the containment to a long term heat sink.

2. The containment boundary must be protected and functional.

3. The containment and reactor coolant system conditions must be well below the point where chemical or physical processes (severe accident phenomena) might result in a dynamic change in containment conditions or a failure of the containment boundary.

The first two of these conditions are relatively straight-forward for the AP600. The energy removal condition requires that a heat sink be available and that a process for getting the energy from the containment to the heat sink exists. Without a means to remove the energy transferred from the core and from chemical processes occurring during a severe accident, the containment pressure and/or temperature will increase to the point where the containment structural integrity could be challenged. Thus, ensuring that an adequate containment heat sink exists will prevent containment pressures and temperatures from reaching the point where the integrity of the containment boundary is challenged. For the AP600 plant, the primary containment cooling mechanism is the Passive Containment Cooling System (PCCS). This system causes the gravity drain of water onto the outside of the steel containment vessel, which then evaporates into the natural circulation air flow around the containment vessel. Even without the water, air cooling alone is sufficient to remove decay heat that has been transferred to the containment. In this scenario, the containment remains below the failure pressure, although it will exceed the design pressure. In addition, the AP600 fan cooler system may be available to supplement the passive cooling. However, the AP600 fan cooler by itself would not be able to keep up with the decay heat production.

The containment boundary condition requires that containment isolation be established and maintained. In the case of severe accidents the containment boundary includes <u>all</u> piping which penetrates the containment and which can have an unrestricted pathway to the environment. These pipes can be considered to be isolated if at least one valve in the pipe is closed (plus any bypass valves in parallel pipes), the line is pressurized with water, or a water seal is established in the line. In other words, all piping which is not actively carrying water to or from the containment, as part of severe accident management, must be isolated by closing at least one valve or establishing a sufficient water seal to prevent release of reactor coolant or containment fluids. Containment isolation considerations extend to the steam generators, main steam lines and feedwater lines since steam generator tube faults (either tube failures or pre-existing leaks) are a major concern for severe accident management. The limited number of containment penetrations in the AP600 design greatly simplifies this consideration, compared with the current generation of plants.

The third condition for a controlled, stable containment state is more difficult to accomplish than the previous two conditions. The changes in containment conditions which can lead to challenges to the containment include dynamic changes which cannot be predicted by trending containment parameters and longer term changes which can be more readily predicted by monitoring containment parameters. Both of these types of changes are of interest since they contribute to the potential for failure of the containment boundary. The dynamic and long term changes in containment conditions considered here are a result of severe accident phenomena. The severe accident phenomena considered in this goal decomposition include:

- Hydrogen flammability
- Core/concrete interactions (CCI)
- High pressure melt ejection (HPME), which includes
  - Direct containment heating (DCH)
  - Reactor vessel lift-off
- Steam explosions
- Creep rupture failure of reactor vessel or SG tubes
- Vacuum caused by hydrogen burning or venting

Although the treatment of severe accident phenomenology for AP600 has been addressed in WCAP-13388, the discussions below summarize the impacts on management of the severe accident.

4.2.1 Hydrogen Flammability

The first of the severe accident phenomena to be considered is hydrogen. The containment pressure rise when a flammable hydrogen mixture is burned in containment is a direct function of the mass of hydrogen present in the containment. During a severe accident, hydrogen is expected in the containment as a result of the in-vessel reactions between the fuel rod cladding and the steam as the core overheats. For any accident sequences in which the RCS pressure is low during core melting, most of the hydrogen generated would be released from the RCS to the containment. However, for sequences with high RCS pressures, a large fraction of the in-vessel hydrogen generation might be trapped in the reactor coolant system. For these latter sequences, a failure of the RCS or an intentional action to depressurize the RCS to the containment (such as any stage of the AP600 ADS) after significant core damage has occurred can suddenly change the containment conditions and may have an impact on hydrogen flammability.

Although the AP600 plant is equipped with hydrogen igniters, this discussion is in relationship to scenarios in which the igniters fail and hydrogen accumulates. AP600 analyses have shown that the containment can withstand the pressure transient from the deflagration of the hydrogen equivalent to 100% of cladding oxidation. If significant core debris is released to a dry containment, core/concrete interactions can result in additional hydrogen generation along with carbon monoxide, which is a flammable gas. Another significant source of hydrogen to be considered is from interactions between unreacted (unoxidized) metals in the core debris and water or steam in the containment after reactor vessel failure.

Since a hydrogen burn can result in a change in containment conditions, a controlled, stable containment can only be achieved if the hydrogen is maintained in a nonflammable state and no significant sources of additional hydrogen are expected. Thus, a controlled, stable containment state with respect to flammable gases requires that: a) the core is covered by water, b) the containment hydrogen is less than the global flammability limits for containment conditions near ambient, c) there are no ongoing core concrete interactions, and d) the reactor coolant system is at a low pressure.

4.2.2 Core/Concrete Interaction

Core/concrete interaction (CCI) can produce substantial changes in the containment conditions in a number of different ways. CCI results in the erosion of the bottom of the containment structure and can result in a containment failure at the basemat. CCI also results in the production of hydrogen and carbon monoxide gases which increases the flammable gas concentration in the containment. CCI without an overlying water layer also results in substantial heating of the containment gases via high temperature gas generation, convective heating of existing gases and radiative heating of nearby structures. In addition, CCI can result in core material configurations which may not be readily coolable, even in the presence of an overlying water cover.

Core/concrete interaction can be prevented by having an adequate level of water covering the containment and the reactor cavity floor. The reactor cavity water inventory may also submerge the reactor vessel and thereby prevent the core debris from leaving the reactor vessel. In the event of reactor vessel failure, the water inventory in the reactor cavity can quench and cool core debris in this region to prevent core/concrete interaction. Thus, a controlled, stable containment state, with respect to core/concrete interaction, requires that either: a) the core is in the reactor vessel or b) the containment and reactor cavity floor regions are covered with sufficient water to quench any core debris discharged from the reactor vessel and c) water recirculation back to the cavity is available to maintain the cavity water level, and thus core debris cooling.

4.2.3 High Pressure Melt Ejection

If the reactor vessel fails while the reactor coolant system is at a high pressure, several severe accident phenomena can occur which have the potential for producing substantial changes in the containment conditions. The subsequent high pressure melt ejection (HPME) can produce direct containment heating (DCH) effects which may substantially change the containment pressure and temperature. HPME can also result in vertical movement of the reactor vessel due to the thrust forces generated by core debris escaping through the failure location in the RPV bottom head. Some studies have indicated that the movement of the RPV may result in sufficient movement in other piping connected to the RCS to tear containment penetrations, thereby challenging containment integrity conditions. HPME also produces a substantial change in the containment hydrogen concentration as described under the hydrogen flammability discussion above. HPME is prevented by either preventing reactor vessel failure or by reducing the RCS pressure.

4.2.4 Steam Explosions

Steam explosions, both within the RPV and in the containment, have been postulated as a concern because they may result in substantial changes in containment conditions by creating breaches in the containment boundary. Steam explosions are a subset of core-coolant interactions which can produce rapid pressure changes in the RCS and the containment. Steam explosions have an accompanying shock wave which, by itself can cause damage to the containment or RCS boundary. [Ref. 9]

An evaluation specific to the AP600 design was conducted to investigate the potential for in-vessel steam explosions. The evaluation concludes that in-vessel steam explosions cannot generate sufficient energy, in a short time scale, to generate a missile that could fail the AP600 containment. [Ref. 7] In addition, it was shown that the peak pressure from any potential in-vessel steam explosion is well within the normal operating pressure of the reactor coolant system. Therefore, the integrity of the RCS pressure boundary is not threatened.

Because of the AP600 containment layout, a significant ex-vessel steam explosion from core debris-water [redacted] interaction can occur only in the reactor cavity. Evaluation of both steam generation rates and potential shock waves induced by debris-water interactions shows that their magnitude is not expected to be sufficient to threaten the AP600 containment integrity. [Ref. 7] The impact of the shock wave on the cavity wall and vessel support structure was also evaluated, with the conclusion that the structural integrity of the cavity walls is not threatened. Therefore, the principal consequence of ex-vessel explosive debris-water interaction is to rapidly cool the debris and pressurize the containment. Neither the steam generation nor the shock waves are expected to challenge the containment integrity for any credible accident scenario.

4.2.5 Creep Rupture Failure

Core damage accident scenarios in which the core material is located within the reactor vessel can lead [redacted] to substantial changes in the containment conditions if either the reactor vessel, the reactor coolant system piping or the steam generator tubes should fail. Reactor vessel failure is primarily a result of contact between molten core material and the inside surface of the vessel bottom head. Reactor coolant system piping and steam generator tube failures are primarily a result of the circulation of high temperature gases within the reactor coolant system which leads to creep rupture failure of the piping.

Creep rupture failure of the RCS piping can result in substantial changes in the containment pressure, hydrogen concentration and fission product inventory. Creep rupture failure of the RCS piping can only occur if the RCS pressure is near its nominal operating value and is a result of heating the pipe walls to a high temperature under high stress conditions. Thus, creep rupture failure of the RCS piping can be prevented by reducing the RCS pressure or submerging the RCS piping in water. Creep rupture failure of the SG tubing can result in substantial changes in the containment integrity since the secondary side of the steam generator pressurizes to the SG safety valve setpoint. This creates a direct pathway for fission product transport from the RCS to the environment. Creep rupture failure of the SG tubes is a result of heating the tube walls to a high temperature and can only occur under conditions of high RCS pressure and a dry steam generator secondary side. Thus, creep rupture failure of the SG tubes can be prevented by reducing the RCS pressure or maintaining an adequate SG secondary side water inventory.

4.2.6 Containment Vacuum

The final severe accident phenomena which must be considered in the definition of a controlled, stable [redacted] containment state is the potential for changes in containment conditions which would result in a substantial vacuum in containment. A substantial vacuum in containment could result in containment boundary failure. These conditions are most likely to be a concern following a large hydrogen burn in the containment or following relief of some portion of the containment gases to the environment. A hydrogen burn will consume some of the oxygen which was present in the containment prior to the accident. Upon condensation of all of the steam in containment and the reduction in containment temperature to near its pre-accident value, the gas volume may be reduced by as much as 21% (assuming all of the oxygen is consumed in a hydrogen burn). This could result in a containment vacuum which challenges the negative design pressure of -2.5 psig.

In severe accident scenarios where a portion of the containment gases were released to the environment, either through late containment isolation or intentional containment venting, the potential for a strong containment vacuum which threatens containment integrity may also exist. To prevent these conditions, air or water must be introduced to the containment such that the containment pressure is within the normal range when the containment temperature is near its nominal value. Thus, a controlled, stable containment state requires that the containment pressure be nearly ambient with no further significant decreases expected.

To maximize the possibility of achieving a controlled, stable containment condition, a summary of all the elements that must be considered in severe accident management are:

- heat transfer from the containment,
- isolation of containment,
- hydrogen prevention/control,
- core/concrete interaction prevention
- high pressure melt ejection prevention,
- creep rupture prevention, and
- containment vacuum prevention.

4.3 Fission Product Release Prevention, Termination and Mitigation

To achieve the goal of terminating fission product releases from the plant, several conditions must be met:

1. The isolation of the containment boundary, including penetrations and steam generator tubes, must be maintained.

2. The fission product inventory of the containment atmosphere must be minimized.

3. Significant leakage through the containment boundary must be stopped.

Some of these conditions may be duplicates of previous conditions for maintaining a controlled, stable core and/or containment state. They are also included here to reinforce the goal of controlling and terminating fission product releases during a severe accident.

Prevention (or termination) of fission product releases therefore requires that the containment boundary be maintained and/or isolated or that the driving force for leakage be eliminated. The containment boundary includes the containment structure, the containment penetrations, the steam generators tubes, and the piping of systems connected to the RCS or containment up to the first isolation valve which is operable. Isolation of the containment boundary includes: a) maintaining existing containment boundaries, and b) closing appropriate valves that isolate systems directly connected to the containment atmosphere or the reactor coolant system, or c) creating a water seal whose static head is greater than the driving force where the first two methods are not available. All of the considerations for maintaining the containment boundary to prevent uncontrolled fission product releases are covered under the goal of maintaining a controlled, stable containment state and are not repeated for the termination and mitigation of fission product releases.

Reducing the inventory of fission products available for release can be a function of the release pathway, which may be directly tied to the accident sequence. For containment releases, the fission product inventory airborne in the containment can be reduced by maintaining the RCS integrity thereby retaining a large fraction of fission products in the RCS. In addition, flooding the containment to submerge RCS piping and flooding the steam generators to submerge the U-tubes would provide cold surfaces for fission product deposition and retention. For release pathways which bypass containment, such as steam generator tube faults or leaks and interfacing system LOCAs (ISLOCAs), the fission product transport can be reduced by reducing the RCS pressure. Also, fission product scrubbing by submerging the release pathway is effective in reducing the dispersion.

In the AP600 design, airborne fission product removal is performed by the operation of the passive containment cooling system. The steam released in containment is condensed on the steel containment shell due to cooling from the passive containment cooling system and this process removes airborne fission products. Fission products that are deposited in the containment sump are retained within the water by the containment sump pH control system. This system adds sodium hydroxide to the floodup inventory in containment to maintain the required recirculation sump pH to promote fission product retention.

The final condition for this goal is to actually terminate the leakage from the containment. Terminating leakage includes eliminating the driving force for leakage (generally a pressure differential across a leakage path), isolating the leaking system, or creating a water seal whose static head is greater than the driving force. Several sources of leakage are worthy of consideration in the SAMG, including: containment, steam generators, and systems connected to either the containment or the RCS. Low levels of leakage from these sources are permitted within the plant design basis. The results of analyses which establish permissible leakage, with respect to offsite doses, are reported in the plant Safety Analysis Report. However, to de-escalate the emergency condition during a severe accident, essentially all of the leakage must be terminated. In the case of containment sources, the leakage can be terminated by reducing containment pressure to near atmospheric. Leakage through containment penetrations can be terminated by closing all valves in the piping and/or by creating water seals in the piping. In the case of the steam generator tubes, leakage can be terminated by keeping the secondary system pressure above the RCS pressure. For systems connected to the RCS and containment, leakage can be stopped by finding alternative methods of accomplishing the same function. For example, recirculation systems which involve transporting high levels of radioactive water outside the containment can result in even very small amounts of leakage being significant. Use of systems that keep all radioactive water within the containment are preferred.

To maximize the possibility of terminating fission product releases, a summary of all the elements that must be considered in severe accident management are:

- isolation of containment,
- reduce fission product inventory, and
- reduce fission product driving force.

4.4 Secondary Goals

Although the previous sections have addressed the goals that must be met for successful mitigation of [redacted] severe accident, there are two additional considerations that should be addressed. These considerations have been termed "secondary goals," since they are not fundamental to the termination of the accident, but their impact is widespread. The secondary goals affect the evaluation of which actions, or strategies, to implement. The two secondary goals are: i) to minimize fission product releases, and ii) to maximize equipment and monitoring capabilities.

The secondary goal to minimize fission product releases is similar to the primary goal of terminating fission product releases. However, the distinction is in the recognition that there may be a need to create an intentional, controlled, and short term fission product release to prevent a larger, uncontrolled, and long term release. Specifically, this is in reference to containment venting, if there is believed to be an immediate threat to the integrity of the containment structure. However, any action which violates the primary goal of terminating fission product releases should be done in a manner that minimizes the release. Another example is the case of steam generator depressurization. There are pathways that blow down directly to the environment, and other pathways (such as through the condenser) that would allow fission products to be scrubbed, and thus dispersion minimized.

The other secondary goal, to maximize equipment and monitoring capabilities, acknowledges that the survivability of some equipment and instruments may become questionable under some severe accident conditions. In general, severe accident conditions are no more severe than the design basis for instrumentation inside the containment. Depending on the scenario, however, temperatures and pressures may exceed the containment design basis, and thus the operability of instruments and equipment is uncertain. Therefore, when making severe accident management decisions, the impact on the instruments and equipment is a factor that should be included in the evaluation process. [redacted]

The capability to repair and maintain equipment following the onset of a severe accident is also important. First, to arrive at a severe accident condition, it is quite likely that some of the plant equipment is not operable. Second, during a severe accident, the potential exists for malfunctions in equipment which is being used during the recovery. Third, since equipment may be used in non-standard ways for severe accident response, local access to areas may be required for valve alignments and/or equipment maintenance. Either the severe accident progression or actions taken to recover from the severe accident conditions may compromise the habitability, particularly due to high radiation levels, of certain plant areas and result in a condition in which some equipment cannot be aligned, maintained or repaired. As in the case of environmental conditions and power supplies for equipment operability, severe accident management decisions should take into account the habitability of plant areas in which alignment, maintenance or repair of equipment enhances the recovery capabilities.

5.0 HIGH LEVEL ACTIONS FOR AP600

Based on the severe accident management goals defined in the previous section, certain elements are necessary to meet the goals. The elements can then further be divided into actions to be taken. The relationship of severe accident management goals to potential actions are summarized in Table 5-1, which forms the basis for possible severe accident management strategies.

The definition of a strategy for AP600 severe accident management consists of three components. A strategy is 1) an action or set of actions that 2) is taken for a specific purpose with 3) specific piece(s) of equipment. A strategy includes more than just the action, since the purposes must be well-understood for an effective evaluation, and the equipment to be used may impact the positive and negative expectations. This is the same strategy definition that was used for the WOG SAMG program, and it initially produced a list of over fifty strategies. Eventually, the strategies were combined to form a smaller number of guidelines, and they were grouped based on the potential actions. Since the AP600 severe accident management program is being developed based on the WOG SAMG program, this same process will be followed.

The information within this section is grouped according to the high level actions that may be taken during the management of a severe accident. The discussion of each high level action includes the identification of the benefits (purposes) of the action, the potential negative impacts, and the equipment possibilities. The development of the high level strategies is a preliminary step in the development of the AP600 severe accident management guidance. The information contained within this section is subject to change based on the completion and finalization of 1) Rev. 0 of the WOG SAMG, 2) the AP600 PRA, 3) AP600 accident management insights and their disposition, and 4) the AP600 function-based task analyses.

5.1 Inject into RCS

Injecting water into the RCS is the most fundamental action to mitigate the progression of a core damage accident. Regardless of where the core has relocated, the RCS may be the most effective pathway to get the water to the core debris. The underlying cause of all severe accidents is the inability to remove the decay heat generated by the core. Therefore, injecting water to the core region is the most direct means of restoring core cooling and stopping the accident progression.

As just stated, one of the benefits of injecting water into the RCS is the restoration of core cooling. The only possibility of preventing the core from relocating to the RPV lower head is to restore injection flow. As water initially flows to an overheated core, the water will flash to steam due to the high temperatures in the core region. Heat can be removed from the core by sensible and latent heat addition to the water and sensible heat addition to the steam. If the flow of water can remove energy at a rate exceeding the decay heat rate, then core cooling can eventually be restored.

Table 5-1
AP600 High Level Actions Relative to Severe Accident Management Goals

Goal	Element	High Level Action
Controlled, stable core	Water Inventory in RCS	- Inject into RCS - Depressurize RCS
	Water Inventory in Containment	- Inject into Containment
	Heat Transfer to SGs	- Inject into RCS - Inject into SGs - Depressurize SGs
	Heat Transfer to Containment	- Inject into RCS - Inject into Containment - Depressurize RCS
Controlled, stable containment	Heat Transfer from Containment	- Depressurize containment - Vent Containment
	Isolation of Containment	- Inject into SGs - Depressurize RCS
	Hydrogen Prevention/Control	- Vent Containment - Pressurize Containment - Burn Hydrogen - Depressurize RCS - Inject into Containment
	CCI Prevention	- Inject into Containment
	HPME Prevention	- Inject into Containment - Depressurize RCS
	Creep Rupture Prevention	- Depressurize RCS - Inject into SGs
	Containment Vacuum Prevention	- Pressurize Containment
Terminate fission product releases	Isolation of Containment	- Inject into SGs - Depressurize RCS
	Reduce Fission Product Inventory	- Inject into Containment - Depressurize RCS
	Reduce F.P. Driving Force	- Depressurize Containment

[redacted]

Another benefit of injecting water into the RCS is the scrubbing of fission products. If a pool of water is overlying a core debris bed, fission products released from the core debris bed will be scrubbed by the water pool. Fission product scrubbing can result in a significant reduction in the amount of fission products released to the containment atmosphere. A water depth of a few feet is a sufficient level to significantly increase the decontamination factor. [Ref. 10] a,c

Finally, injection of water into the RCS may help retain the core within the reactor vessel. The energy removed by the water can slow the core damage progression and may delay or even prevent vessel failure. The injection of water during the TMI-2 accident, for example, provided sufficient heat removal to retain the core debris within the vessel. However, there is no guarantee that the injection of water in another severe accident would prevent the vessel from failing. Nevertheless, even a delay in vessel failure is a benefit worth achieving.

There are also negative impacts from injecting water onto hot core debris during a severe accident. The potential impact of these adverse effects should be considered before a decision is made to implement the strategy. The potential negative impacts are the production of hydrogen and the potential for creep rupture of the steam generator tubes. A summary of these drawbacks is provided below.

The hot fuel cladding, in the presence of steam, oxidizes and produces significant amounts of hydrogen. If the containment hydrogen igniters are not working, the accumulation of hydrogen in the containment is a concern. Although AP600 analyses have shown that the containment can withstand the deflagration of hydrogen produced from 100% of the cladding being oxidized, the containment integrity could be challenged if there are significant additional combustible gases. The production of hydrogen is unavoidable when adding water to an overheated core (above 1800°F). However, the total hydrogen production for accidents where the core is recovered in-vessel should be much less than 100% cladding oxidation. Ultimately, to achieve a controlled, stable containment, the possibility of future hydrogen production must be minimized by covering the core with water. Without the reflooding of the core debris, the potential exists for significant additional hydrogen production that could later create a containment challenge. Therefore, although the hydrogen production due to injecting water into the RCS is a negative impact, it is not a containment challenge.

Another potential negative impact is creep rupture of the steam generator tubes. This is a failure mode that can occur when the steam generator tubes are subjected to high temperatures and large primary-to-secondary pressure differences. Tube temperatures can reach creep rupture limits quickly if hot gases that accumulate in the core upper plenum are forced into the steam generators by the rapid steaming that will occur when injection into the RCS reaches the overheated core debris in the reactor vessel. Since the steam generator tubes provide a fission product boundary, maintaining the tube integrity during a severe accident is important to the goal of eliminating fission product releases. There are two methods of preventing these adverse impacts: decrease the primary-to-secondary pressure difference, or make sure that the steam generator tubes are at least partially covered on the secondary side.

Two negative impacts that were also included in the WOG SAMG but that are not applicable to the AP600 design are containment flooding and an insufficient injection source. Containment flooding for current plants is a concern because equipment such as containment vent valves or fan cooler exhaust/intake ducts may be located where they will be covered with water and unusable. However, the AP600 plant has been designed so that significant containment flooding does not affect necessary equipment. Another concern in most current plants is that the use of the water from the RWST may limit other uses of that water. However, the AP600 plant is designed so that there are rarely systems competing for the same water inventory, and thus the injection of water does not impact the ability to perform other actions.

Furthermore, the design of the injection systems for the AP600 plant is significantly different than existing plants. Current plants rely primarily on the forced injection of water from sources outside the containment. However, the AP600 plant relies primarily on the passive injection from water tanks inside the containment. Each of the water tanks and injection methods for AP600 is further discussed below.

The Core Makeup Tank (CMT) is a safety-related means to provide water inventory to the RCS. There are two CMTs, each with a capacity of 2000 ft^3, which replace the function of high head safety injection pumps in current plants. CMTs are located above the reactor coolant loops and have two pressure balancing lines: one from the top of the pressurizer and another from a cold leg pipe. The CMTs are maintained full of borated water and are designed to inject at any RCS pressure. The discharge from the CMTs is routed from the bottom of the tanks to separate safety injection nozzles on the reactor vessel. Each discharge line is normally isolated by two parallel air-operated valves that fail open on loss of air pressure, loss of dc power, or loss of control signal.

The AP600 is also provided with two accumulators that supply borated water at high makeup flow rates to refill the reactor vessel downcomer and lower plenum during a large loss of coolant accident or during other events requiring automatic or manual RCS depressurization. The back pressure for the accumulators is 700 psig, so that the RCS pressure must be reduced below this value before water will inject.

The in-containment refueling water storage tank (IRWST), in conjunction with the automatic depressurization system, provides the function of low head safety injection. To get injection from the in-containment refueling water storage tank, the RCS pressure must be reduced to a value near containment pressure. The automatic depressurization system is provided to accomplish this function. When the IRWST empties, the containment is flooded above the RCS loop level, and the water in the containment drains, by gravity, back into the RCS if there is a break in the hot or cold leg. Therefore, stable, long-term core cooling and makeup to the RCS is established. The passive containment cooling system supports this operation by removing heat from the containment. Steam released from the RCS is condensed. This condensate then drains back into the RCS for recirculation.

The normal residual heat removal system (NRHR) provides an additional mechanism for core cooling, taking water from the IRWST/sump and injecting it into the safety injection lines. The NRHR system needs cooling water and ac electrical power. If offsite power is lost, the power is supplied by two non-safety-related diesel generators. The NRHR loops take water outside the containment, which could be a negative factor during a severe accident, since the coolant water is highly contaminated with fission products and may result in more personnel access restrictions.

Finally, the chemical and volume control system (CVS) is the normal RCS inventory makeup system. It has two non-safety grade high pressure pumps, which start automatically if a core makeup tank actuation signal is generated. The pumps are also automatically loaded on the non-safety diesel generators if offsite power is lost. The CVS is rated to provide a flowrate around 100 gpm per pump at full system pressure. If the core is totally uncovered, the CVS is insufficient, by itself, to quench the core and reflood the vessel. However, this may be sufficient to remove decay heat as it is produced.

Because the AP600 plant is a passive design, most of the methods of injecting water rely on the gravity draining of tanks. However, for this to occur from some of the tanks, the RCS must be depressurized. The largest tank, the IRWST, requires that the RCS be almost fully depressurized. Therefore, during a severe accident in which injection capability has failed, it may be due to the RCS having a pressure that is too high. This makes the action of depressurizing the RCS very important for AP600. This high level action is further discussed in Section 5.4.

5.2 Inject Into Containment

Another important strategy for AP600 is to inject water into the containment cavity so that water surrounds the outside of the reactor vessel. According to WCAP-13388 [Ref. 7], this action has more impact on accident management considerations than any other individual phenomena except the direct water addition to the debris. Based on experiments and analyses, it is believed that external flooding of the vessel can cause the core debris to be retained within the lower vessel plenum. This action also has the benefits of protecting the containment, creating a heat removal path from the core debris, stopping the accident progression through the prevention of vessel failure, and preventing all ex-vessel phenomena from occurring.

However, if the core is ex-vessel or if vessel failure is imminent, the injection of water into the containment has other benefits. The presence of water in the cavity will scrub fission product inventory, eliminate high pressure melt ejection concerns, and will prevent or limit core/concrete interaction (CCI). CCI is the phenomena of core debris attacking the concrete basemat if there is no water in the containment cavity to cool the debris. The consequences of CCI include the generation of non-condensable gases that will pressurize the containment, the generation of combustible gases that can ignite and fail the containment, the generation of a significant amount of aerosols, and the eventual failure of the containment boundary due to basemat or liner melt-through.

Injecting water into the containment to a level that covers the RCS loops is a viable action for the AP600 design and automatically occurs if all in-containment water sources are directed into the containment cavity. For core damage events resulting from a large LOCA, this containment water level will ensure that water gets into the reactor vessel. For non-LOCA events, this water level protects RCS loops from creep failure and cools gases to prevent steam generator tube creep rupture.

There are very few negative considerations of injecting water into the containment cavity. In the WOG SAMG, three negative impacts are identified, which are 1) de-inerting the containment if the sprays are used, 2) using the water inventory in the RWST that may be needed for other actions, and 3) pressurizing the containment to the point that gravity drain of the RWST would not be possible. For AP600, none of these negative impacts apply. The AP600 containment design does not include any internal containment sprays. There are also no competing uses for the IRWST water. And if the method of containment injection is gravity drain of the IRWST, the flow rate is high and is possible regardless of containment pressure. This is because the IRWST is internal to the containment and gravity drain would not be impacted by containment pressure.

Another potential negative impact that was discussed in the WOG SAMG and is also applicable to AP600 is the potential for an ex-vessel steam explosion if the vessel fails. A steam explosion could result in the destruction of the reactor cavity walls which provide support for the reactor vessel. If a steam explosion destroys the reactor vessel supports, the containment fission product boundary may be challenged due to tearing of containment penetrations connected to the RCS as the reactor vessel drops to the reactor cavity floor. Evaluations of the AP600 reactor cavity wall structural capability concludes that steam explosions in the reactor cavity do not pose a challenge to the containment fission product boundaries. The cavity walls are expected to withstand the effects of any credible steam explosion.

5.3 Inject Into Steam Generators

In conventional plants, the steam generators are designed to provide a heat sink for the RCS during both normal and accident conditions. Therefore, in the WOG SAMG, injecting into the steam generators was judged to be one of the most important activities. In the AP600 plant, although the steam generators are designed for heat removal during normal operation, they are not a safety-related method of decay heat removal during an accident. However, injecting into the steam generators is still an important high level action for the management of a severe accident in the AP600 plant.

Because much of the secondary side is located outside of containment, the SG tubes act as a containment boundary. Therefore, the prevention of induced steam generator tube ruptures is important for severe accident management. One of the methods of doing this is to inject water into the steam generator to keep the tubes cooled. This protects them from rupturing due to heatup from hot gases on the primary side of the tubes. Nevertheless, if a tube rupture does occur, covering the break with water will scrub fission products from the primary system following core damage. a,c

Also, although the AP600 steam generators are not a safety-related method for removing decay heat, they may still be useful for this function during a severe accident. Not only may the heat removal be of benefit, but the steam generators may be a method for depressurizing the RCS.

However, there are also several drawbacks associated with injecting water into the steam generators. These drawbacks have the potential to negatively impact the accident progression by allowing the direct release of fission products to the environment. The first concern is the thermal shock of the steam generators. If the steam generators have dried out during a severe accident, the tube temperatures may exceed 1000°F. The injection of cold water to the hot, dry steam generators can place significant thermal stresses on the tubes and other components. These thermal stresses can result in the failure of either the shell side of the steam generator or the steam generator tubes. Failure of the shell side of a steam generator during a severe accident reduces the amount of water that can enter the steam generator and increases flooding of the containment. Failure of one or more tubes will result in a containment bypass and the potential release of fission products to the environment.

Another potential concern with the injection of water into the steam generators can occur if the steam generators must first be depressurized. If the RCS is pressurized, the depressurization of the steam generators could create a large primary-to-secondary ΔP that could induce creep rupture of the steam generator tubes. Either this induced tube rupture, or pre-existing tube ruptures, would then make fission product releases to the environment a concern. These potential negative impacts must be considered in the evaluation process that determines whether steam generator injection should be attempted.

Equipment that may be used for this high level action is dependent on the pressure of the steam generators. For high pressures, only the startup feedwater pumps and the main feedwater pumps would be usable. For lower pressures, the list of possible equipment expands to include condensate pumps, firewater pumps and service water pumps.

5.4 Depressurize RCS

Many benefits can be realized by depressurizing the RCS during a severe accident. As previously discussed in Section 5.1, the depressurization of the RCS will facilitate the injection of water from the passive core cooling system tanks. It will also increase the flowrate that would be provided if pumps are being used. If the ADS valves are used, the creation of the intentional opening in the RCS may be the method of establishing a long term heat removal path.

There are also many other effects of depressurizing the RCS that are unique to core damage scenarios. The possibility of creep rupture of the steam generator tubes and the RCS pipes can be reduced or eliminated if the RCS pressure is lowered. Creep rupture is a plastic deformation process that occurs under high temperatures and sustained loads. Since high temperatures are a by-product of the severe accident, the reduction of the RCS pressure is a good method to avoid failures due to creep rupture.

Another important severe accident concern is high pressure melt ejection (HPME). This is a phenomenon that may occur if the RCS pressure is elevated at the time of vessel failure. During HPME, the momentum of the core debris along with the driving force of high velocity gases released from the vessel, can transport the molten core debris away from the reactor cavity region. One method of preventing this phenomenon is to decrease the RCS pressure.

Decreasing the RCS pressure also can help isolate the containment and reduce fission product releases for containment bypass sequences. If there are ruptures or leaks in the steam generator tubes, the reduction of the RCS pressure will reduce the driving force on the fission products, and will help to maintain them within the primary system. In addition, if injection of water occurs due to the reduction in RCS pressure, the water inventory will help to scrub the fission products.

The final benefit of reducing the RCS pressure is for long term control of the hydrogen inventory. In order to exit the severe accident management guidance, the containment must be in a controlled, stable state. Part of the definition of this state is that there should be no potential for sudden, future changes to the containment atmosphere. If the RCS remains pressurized with hydrogen accumulated within the system, any future failure of the vessel or opening in the RCS would release the hydrogen to the containment atmosphere. Therefore, the RCS pressure should be reduced for long term concerns.

However, the long term benefit of hydrogen control also produces a short term negative impact. If the RCS is depressurized using a vent path to the containment, the sudden release of a large quantity of hydrogen to the containment could change the flammability status of the containment atmosphere. Although AP600 analyses have shown that the containment structure can withstand the resulting pressure transient, the plant decision-makers should be aware of the potential of the burn. Also with the opening of a pathway from the RCS to the containment, there could be a sudden increase in the containment pressure. This pressure increase is not of sufficient magnitude to challenge the containment integrity.

The safety grade system for depressurizing the RCS is the Automatic Depressurization System (ADS). This system is a series of valves arranged in four stages, which provide a phased depressurization capability. The valves of the first three stages are motor-operated valves and are mounted on the pressurizer. These valves discharge steam to the IRWST through spargers. The discharged steam is condensed and cooled by mixing it with water in the tank. The valves of the fourth stage are air-operated valves and are located on lines connected to the two hot leg pipes. The fourth stage vents directly to containment.

Other equipment for depressurizing the RCS that will be investigated are the pressurizer spray, the RCS head vent, and CVS letdown. Another method is to depressurize the RCS via heat removal from the steam generators. The equipment associated with this action will be addressed in the following section.

5.5 Depressurize Steam Generators

The purposes of this high level action have been discussed in Sections 5.3 and 5.4. Depressurizing the steam generators may be the first step to enable injection of water into the SGs, to establish a heat transfer path from the RCS to the SGs, or to depressurize the RCS. The end purpose of this action may be the depressurization of the RCS, or the establishment of a long term decay heat removal pathway.

The principal negative impacts from depressurizing the steam generators are related to the potential for creating a release pathway. Not only might the steam generator inventory be lessened, but any fission products within the steam generators may be released to the environment. Furthermore, if there is a steam generator tube rupture, the lower steam generator pressure will increase the driving force of fission products from the primary to the secondary side. Even if no steam generator tube ruptures currently exist, the lowering of the steam generator pressure could increase the ΔP across the steam generator tubes, inducing a rupture or increasing leakage of fission products from the RCS through leaking steam generator tubes.

The two principal methods of depressurizing the steam generators are opening the SG power operated relief valves which discharge directly to environment or opening the steam dump valves which discharge to the main condenser. There are no known differences in the AP600 design, when compared to existing Westinghouse PWR plants, that would impact the equipment to perform this high level action.

5.6 Depressurize Containment

During a severe accident it is likely that the containment will experience a substantial increase in pressure. Unless the RCS is intact, and the steam generators are being used for heat removal, all of the energy generated during the accident must ultimately be removed through the containment. Until the energy is effectively removed, the containment will pressurize. Therefore, one of the high level actions for severe accident management is to depressurize the containment.

There are several benefits to depressurizing the containment. The fundamental benefit, as mentioned above, is the ultimate heat removal from the core, which is needed to conclude that the plant is in a controlled, stable state. However, depressurizing the containment might also be needed for immediate concerns. If overpressurization threatens the integrity of the containment, depressurization would be needed to address this severe challenge. Also, depressurization may be the method of reducing or eliminating fission product releases from the containment. Reducing the ΔP from the containment to the environment reduces the driving force behind the fission product leakage. Another benefit of depressurizing the containment is the general improvement of the containment atmosphere to alleviate potential equipment and instrumentation challenges. Finally, depressurizing the containment by condensing steam will increase water in the IRWST and containment for ex-vessel core debris cooling and flooding of the reactor protection vessel and RCS loops.

The potential negative impact from depressurizing the containment is that with the condensation of steam from the containment atmosphere, the hydrogen becomes a larger fraction of the overall containment atmosphere. The higher hydrogen fraction may lead to a flammable state inside the containment. If the hydrogen was previously inflammable at the higher pressure due to the presence of steam, and if it becomes flammable due to the condensation of steam, this process is known as de-inerting. However, this concern is not anticipated for AP600 due to the existence of hydrogen igniters. Nevertheless, if the igniters were not functioning properly and significant hydrogen accumulated, the containment could be de-inerted by depressurization. Thus, the containment boundary could be challenged.

The AP600 passive safety-related containment cooling is provided by a water tank that allows a gravity-fed flow onto the outside of the containment dome surface, with sufficient water for three days. After three days, heat can be removed by convective air flow to maintain containment pressure below design pressure. There are also two fan coolers, although they are designed for normal operation and thus their heat removal capacity is low. The final option to depressurize the containment is to vent the containment, which is addressed as a separate high level action in Section 5.9.

5.7 Pressurize Containment

This high level action addresses two very different concerns. The first concern is to pressurize the containment to create a steam inert atmosphere that would prevent a hydrogen burn. As discussed in the previous section, the presence of a sufficient quantity of steam in the containment atmosphere can ensure that the hydrogen is not flammable, and thus the containment is "inert." However, pressurizing the containment to create an inert containment atmosphere is only a temporary solution. The passive features of the AP600 containment cooling system will also be working to condense steam, and the removal of the hydrogen will eventually be needed.

On the other end of the spectrum, the high level action of pressurizing the containment is to prevent a vacuum failure of the containment due to too low of a pressure. The threat of a containment vacuum could be created by previous containment venting, delayed containment isolation, or hydrogen burns that have substantially reduced the oxygen in the containment atmosphere.

The methods suggested in the WOG SAMG to accomplish these actions focus on turning off the containment heat sinks. For AP600 the gravity drain of the water over the outside of the containment shell may be terminated, which will lessen the heat transfer. Another pressurization method in the WOG SAMG is to open a pressurizer power-operated relief valve, if the RCS is still intact, to release steam into the containment. This method is also available for the AP600 plant via the 4th stage ADS valves. In addition, the IRWST, if it is being used via the PRHR system, steams directly to the containment.

If containment pressurization is being performed to prevent a containment vacuum, another option is to introduce instrument air into the containment. However, the negative impacts of this action are that there will be more oxygen that could be used in a hydrogen burn, and the possible failure to isolate the path being used for pressurization.

5.8 Burn Hydrogen

If hydrogen igniters are not functioning properly, it may be desirable to intentionally burn the hydrogen using other methods to create the initial spark. If the containment atmosphere is flammable, it is possible that an immediate smaller burn may be preferable to a larger burn later. Since the AP600 containment is believed to be capable of withstanding a hydrogen burn from all the fuel cladding being oxidized, this action may only become a factor when CCI is believed to be a potential concern. The negative impact would be a brief temperature and pressure spike in the containment. The methods that may be successful at creating the needed spark will be investigated during a later phase of the development of AP600 severe accident management guidance. One of the options that will be considered is to establish an alternate power source to the hydrogen igniters.

5.9 Vent Containment

Venting the containment is the last high level action to be addressed since the negative impacts from implementing this action are relatively certain. However, there are two strategies that consider this action as a method of achieving the long term goals of severe accident management. The first reason to consider venting is if the containment pressure has increased to the point that failure of the containment pressure boundary is expected. If the accident sequence has resulted in more severe containment conditions than anticipated, and if the heat sinks have not functioned as expected, there could be a need to consider the intentional venting of the containment. This would result in a release of fission products to the environment. But a short term release from which control can be regained may be preferable to a large release as a result of the failure of the containment structure.

Another reason to consider venting is as a hydrogen control measure in the containment. If hydrogen igniters have not functioned properly, and if core/concrete interaction has contributed to the hydrogen inventory, the containment integrity may be threatened by the potential for a hydrogen burn. Less drastic hydrogen control measures include inducing a hydrogen burn while the concentration is low enough that the possibility of containment failure can be precluded (Section 5.8) and pressurizing the containment to create an inert atmosphere (Section 5.7). However, the latter option is only a temporary solution, and it may be too late to implement the former option. Therefore, containment venting is also an action that may be considered as a method of hydrogen control. Note that venting the containment does not reduce the flammability of the containment atmosphere, however, it reduces the impact of a hydrogen burn. This is because the containment pressure rise when a flammable hydrogen mixture is burned in containment is a direct function of the mass of hydrogen present in the containment. Therefore, reducing the hydrogen mass will reduce the amount of energy released in a burn. The hydrogen can be reduced, through venting, to the point that there is not enough energy to fail the containment.

The main negative impact of venting the containment is obviously the radiological release of fission products to the environment. Ideally, this release would be relatively small. However, there is also the possibility that the vent pathway cannot be re-isolated. In addition, the release of non-condensable gases during the venting leads to the potential for a future challenge of the containment pressure boundary due to a containment vacuum. If non-condensable gases are released, containment isolation is re-established, and steam condenses from the atmosphere, the resulting containment vacuum could be severe enough to fail the pressure boundary due to a compressive load. The methods that may be used to vent the AP600 containment will be investigated during a later phase of the development of the severe accident management guidance.

[redacted]

TABLE 5-2 SUMMARY OF HIGH LEVEL SEVERE ACCIDENT MANAGEMENT STRATEGIES FOR AP600			
ACTION	**PURPOSE** **(Positive Impacts)**	**OTHER CONSIDERATIONS** **(Negative Impacts)**	**EQUIPMENT**
Inject into RCS	• To restore core cooling (immediate and long term) • To scrub fission products • To prevent, or delay, vessel failure	• Creation of hydrogen • Creep rupture of SG tubes	• CMT • Accumulators • IRWST • NRHR • CVS
Inject into Containment	• Create inventory in sump for recirculation • Cover lower head of RPV to prevent failure • Cool core debris • Prevent/limit CCI - Prevent basemat melt-thru - Reduce flammable gas production • Prevent HPME • Reduce fission product inventory	• Ex-vessel steam explosions	• Gravity Drain of IRWST • SFS injection into refueling cavity
Inject into SGs	• Heat Sink • Cover SG tubes to prevent creep rupture • Scrub fission products • To make SGs available to depressurize RCS	• Thermal shock of SG tubes • F.P. release from leaking tubes • Creep rupture of SG tubes (if SG is first depressurized, creating large ΔP)	High Pressure: - Main FW - Startup FW Low Pressure: - Condensate - Firewater - Service Water
Depressurize RCS	• To facilitate injection into the RCS • To establish long term heat transfer path • To prevent HPME • Prevent creep rupture • Isolate containment due to SG tube leaks • Long term hydrogen control • Reduce fission product inventory	• Short term hydrogen release and burn • Containment pressurization	• ADS • Auxiliary Press'zer Spray • Head Vent • CVS Letdown • via SGs
Depressurize SGs	• To facilitate injection into SGs • To create heat transfer path with RCS • Depressurize RCS	• Loss of SG inventory • SG fission product releases • Creep rupture of SG tubes if large ΔP is created.	• SG PORV • Steam Dump
Depressurize Containment	• Prevent overpressurization • Mitigate containment fission product leakage • Alleviate equipment and instrumentation challenges due to harsh conditions	• Hydrogen flammability • Containment vacuum if venting	• PCCS • Fan Coolers • Vents

a,c

TABLE 5-2 (Continued)
SUMMARY OF HIGH LEVEL SEVERE ACCIDENT MANAGEMENT STRATEGIES FOR AP600

ACTION	PURPOSE (Positive Impacts)	OTHER CONSIDERATIONS (Negative Impacts)	EQUIPMENT
Pressurize Containment	• To create inert atmosphere so that hydrogen cannot burn • To prevent containment vacuum from failing containment structure	• Removal of hydrogen will eventually be needed • More oxygen for hydrogen burn • Possible failure to isolate pathway used for pressurization	• Turning off containment heat sinks: - Fan Coolers - Stop water flow over containment exterior
Intentionally burn hydrogen	• Let hydrogen burn while containment failure is not a risk; to prevent future containment challenge.	• Pressure and temperature spike	• Alternate power source for hydrogen igniters
Vent Containment	• To avoid containment failure due to: - Overpressurization - Hydrogen Burn	• Radiological releases • Potential future concerns with containment failing from sub-atmospheric loads • No guarantee that vent pathway will be able to reclose.	

6.0 CONCLUSION

This document has examined the overall framework for the development of the AP600 severe accident management guidance. The decision-making process, the roles of plant staff, and the high level contents of the guidelines have been examined. In addition, the goals of severe accident management have been defined, and necessary high level actions to achieve the goals have been summarized.

The conclusion of this assessment is that the AP600 high level actions for severe accident management are similar to those developed for the WOG SAMG. However, the differences between the AP600 plant and conventional plants result in some differences in accident management applications, as summarized below.

1) Because AP600 does not have internal containment sprays, there is no potential for rapidly de-inerting the containment. This minimizes the negative impacts to be considered when injecting into the containment and depressurizing the containment.

2) The AP600 plant has been designed to facilitate the flooding of the containment cavity. This creates several benefits.

    a) Flooding the containment to cover the RPV lower head increases the probability that the core debris can be retained in vessel and that ex-vessel phenomena can be avoided.

    b) The flooding level is able to cover the RCS loops, which can prevent creep rupture failure of the RCS piping.

    c) Flooding can be accomplished without endangering vital instruments and equipment.

3) The AP600 containment passive cooling lowers the need for operator action to control the containment pressure and temperature.

4) Due to the high natural circulation flow rates in the AP600 RCS, the actions to inject to the steam generators and depressurize the RCS should have high priority to prevent induced steam generator tube ruptures.

5) Because of the passive nature of the RCS injection systems, depressurizing the RCS should have a high priority to facilitate water injection.

7.0 REFERENCES

1) DE-AC03-90SF18495, AP600 Standard Safety Analysis Report, Section 18.9.8.1, "Development of the Emergency Operating Procedures"

2) SECY-93-308, "Status of Implementation Plan for Closure of Severe Accident Issues, Status of the Individual Plant Examinations and Status of Severe Accident Research," November 15, 1993.

3) NP-6780-L, Advanced Light Water Reactor Utility Requirements Document, Volume III (ALWR Passive Plant), Chapter 1, "Overall Requirements," paragraph 2.3.3.9

4) NRC Project No. 669, "Issuance of Final Safety Evaluation Report (FSER) on the Electric Power Research Institute (EPRI) Requirements Document for Passive Plant Designs," from R. W. Borchardt, Office of Nuclear Reactor Regulation, August 31, 1993.

5) EPRI TR-101869, Severe Accident Management Guidance Technical Basis Report, Volume 2, Appendix K, "Debris Transport to the Lower Plenum."

6) EPRI TR-101869, Severe Accident Management Guidance Technical Basis Report, Volume 2, Appendix BB, "Potential for Criticality of the Core Material Under Recovery From Severe Accident Conditions."

7) WCAP-13388 (Proprietary) and WCAP-13389 (Non-Proprietary), AP600 Phenomenological Evaluation Summaries.

8) EPRI TR-101869, Severe Accident Management Guidance Technical Basis Report, Volume 2, Appendix L, "External Cooling of the RPV with Debris in the Lower Plenum."

9) EPRI TR-101869, Severe Accident Management Guidance Technical Basis Report, Volume 2, Appendix G, "Steam Explosions."

10) EPRI TR-101869, Severe Accident Management Guidance Technical Basis Report, Volume 2, Appendix U, "Water Overlying Core Debris."

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DCP/NRC1413
NSD-NRC-98-5757
Docket No.: 52-003

August 14, 1998

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

ATTENTION: T. R. Quay

SUBJECT: RESPONSE TO NRC LETTERS CONCERNING REQUEST FOR WITHHOLDING INFORMATION

Reference:

1. Letter, Sebrosky to McIntyre, "Request for withholding information from public disclosure for Westinghouse AP600 design letter of October 20, 1993," dated June 18, 1998.

2. Letter, Sebrosky to McIntyre, "Request for withholding information from public disclosure for Westinghouse AP600 design letter of January 17, 1994," dated June 18, 1998.

3. Letter, Sebrosky to McIntyre, "Request for withholding information from public disclosure for Westinghouse AP600 letters of September 20, 1993, January 21, 1994, and February 3, 1994," dated July 10, 1998.

4. Letter, Sebrosky to McIntyre, "Request for withholding proprietary information for Westinghouse letters dated April 18, 1995," dated July 15, 1998.

5. Letter, Huffman to McIntyre, "Request for withholding information from public disclosure of Westinghouse report on AP600 function based task analysis," dated July 17, 1998.

Dear Mr. Quay:

Reference 1 provided the NRC assessment of the Westinghouse claim that proprietary information was provided in a letter dated October 20, 1993, that contained the response to a staff request for additional information regarding the AP600 probabilistic risk assessment. The NRC assessment was that the material was similar to material that exists in the current (1998) nonproprietary version of the AP600 probabilistic risk assessment (PRA) report. In addition, the staff indicated the material was used by the staff in the development of the AP600 draft safety evaluation report and therefore should remain on the docket. At the time this request for additional information response was provided to the

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Enclosure 2

NRC technical staff, the information was considered to be proprietary by Westinghouse since it contained information that had commercial value to Westinghouse. If this request for additional information response was indeed used by the staff in development of the AP600 draft final safety evaluation report in November 30, 1994, then at this time, almost five years later, this information is no longer considered to be proprietary by Westinghouse.

Reference 2 provided the NRC assessment of the Westinghouse claim that proprietary information was provided in a letter dated January 17, 1994, that contained the response to a staff request for additional information regarding the AP600 instrumentation and control system. The NRC assessment was that the material was similar to material that exists in the current (1998) nonproprietary version of the AP600 standard safety analysis report. In addition, the staff indicated the material was used by the staff in the development of the AP600 draft safety evaluation report and therefore should remain on the docket. At the time this request for additional information response was provided to the NRC technical staff, the information was considered to be proprietary by Westinghouse since it contained information that had commercial value to Westinghouse. If this request for additional information response was indeed used by the staff in development of the AP600 draft final safety evaluation report in November 30, 1994, then at this time, over four years later, this information is no longer considered to be proprietary by Westinghouse.

Reference 3 provided the NRC assessment of the Westinghouse claim that proprietary information was provided in a letter dated September 20, 1993, that contained information related to the AP600 PRA and WCAP-13795, which provided the PRA uncertainty analysis. The NRC assessment was that the material was similar to material that exists in the current (1998) nonproprietary version of the AP600 probabilistic risk assessment (PRA) report. In addition, the staff indicated the material was used by the staff in the development of the AP600 draft safety evaluation report and therefore should remain on the docket. At the time this information was provided to the NRC technical staff, it was considered to be proprietary by Westinghouse since it contained information that had commercial value to Westinghouse. If the information transmitted by the Westinghouse September 20, 1993, letter was indeed used by the staff in development of the AP600 draft final safety evaluation report in November 30, 1994, then at this time, almost five years later, this information is no longer considered to be proprietary by Westinghouse.

Reference 3 also provided the NRC assessment of the Westinghouse claim that proprietary information was provided in a letter dated January 21, 1994, that contained WCAP-13913, "Framework for AP600 Severe Accident Management Guidance" (SAMG). The NRC assessment was that the material was similar to material that exists in current (1998) nonproprietary AP600 documents (e.g., WCAP-13914, "Framework for AP600 Severe Accident Management Guidance"). In addition, the staff indicated the material was used by the staff in the development of the AP600 draft safety evaluation report and therefore should remain on the docket. At the time this Framework for SAMG was provided to the NRC technical staff, the information was considered to be proprietary by Westinghouse since it contained information that had commercial value to Westinghouse. At this time, over four years later, this information is no longer considered to be proprietary by Westinghouse.

Reference 3 also provided the NRC assessment of the Westinghouse claim that proprietary information was provided in a letter dated February 3, 1994, that contained additional copies of WCAP-13913, "Framework for AP600 Severe Accident Management Guidance" (SAMG). The NRC assessment was that the material was similar to material that exists in current (1998) nonproprietary AP600 documents (e.g., WCAP-13914, "Framework for AP600 Severe Accident Management Guidance"). In addition, the staff indicated the material was used by the staff in the development of the AP600 draft safety evaluation report and therefore should remain on the docket. At the time this Framework for SAMG was provided to the NRC technical staff, the information was considered to be proprietary by Westinghouse since it contained information that had commercial value to Westinghouse. At this time, over four years later, this information is no longer considered to be proprietary by Westinghouse.

Reference 4 provided the NRC assessment of the Westinghouse claim that proprietary information was provided in a letter dated April 18, 1995, that contained information for a MAAP4/RELAP comparison for the AP600 in response to a staff request for additional information. The NRC assessment was that the Westinghouse cover letter indicated that Enclosure 2 is a non-proprietary version of Enclosure 3, however, the staff could not find any portion of the enclosures marked as proprietary. The staff assessment further states the conventional bracketed-superscript notation also appears to be missing. Finally, the NRC assessment states the staff could not determine which part of the material enclosed with the Westinghouse letter was Enclosure 1, 2, or 3. It should be noted that the Westinghouse April 18, 1995, cover letter states "Enclosures 2 (nonproprietary) and 3 (proprietary) provide the requested information." The letter does not indicate that enclosure 2 was a duplicate of enclosure 3 minus the proprietary information. A cover sheet was provided just prior to each of the enclosures to the Westinghouse letter. The enclosures contained the following: Enclosure 1 provided a copy of the NRC's two-page request for information for the MAAP-RELAP comparison. Enclosure 2 provided the requested information, and was titled "Requested Information for AP600 MAAP4/RELAP Comparison." Under section 4, Initial Conditions, of Enclosure 2 it states the initial conditions information (which was proprietary) is provided in Enclosure 3 of the subject Westinghouse letter. Finally, Enclosure 3 contained the list of initial conditions. The information provided in Enclosure 3 was labeled as Westinghouse Proprietary Class 2 at the top of the page, however, the specific proprietary information was not indicated by the bracketed-superscripted notation. In addition to the initial conditions, a mark-up of AP600 PRA Figure K-1 was provided in Enclosure 3. Again, the information was labeled as Westinghouse Proprietary Class 2 at the top of the page, however, the specific proprietary information was not indicated by the bracketed-superscripted notation. At the time the information provided in Enclosure 3 of the subject Westinghouse letter was provided to the NRC technical staff, the information was considered to be proprietary by Westinghouse since it contained information that had commercial value to Westinghouse. At this time, over three years later, this information is no longer considered to be proprietary by Westinghouse.

Reference 5 provided the NRC assessment of the Westinghouse claim that proprietary information was provided in a letter dated February 8, 1994, provided a copy of WCAP-13957, "AP600 Reactor Coolant System Mass Inventory: Function Based Risk Analysis." The NRC assessment was that the material was not "information that the staff customarily accepts as proprietary." In addition, the staff indicated the material was used by the staff in the development of the AP600 final safety evaluation report and therefore should remain on the docket. At the time this report was prepared, the information was considered to be proprietary by Westinghouse since it contained information that had commercial value to Westinghouse and was of the type of information that was customarily held in confidence by Westinghouse. That the material was not information that the staff customarily accepts as proprietary is not relevant to making the proprietary determination. However, in an effort to expedite the issuance of the AP600 Final Safety Evaluation Report and Final Design Approval, Westinghouse agrees to no longer consider this information to be proprietary.

In a telephone call on July 8, 1998, the staff informed Westinghouse of a concern related to WCAP-13288 and WCAP-13289, which were associated with the AP600 check valve testing specification. The concern was that the proprietary report had no proprietary information identified and the nonproprietary report had been placed in the public document room. Westinghouse has reviewed these reports and, at this time, considers none of th~ information to be proprietary.

This response addresses the proprietary issues delineated in the references.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

cc: J. W. Roe - NRC/NRR/DRPM
J. M. Sebrosky - NRC/NRR/DRPM
W. C. Huffman - NRC/NRR/DRPM
H. A. Sepp - Westinghouse

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