UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ON MAIN STEAM LINE BREAK - SINGLE FAILURES

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

Consumers Power Company (CPCo), the licensee for the Palisades Plant, committed in 1982 and 1983 to implement certain modifications to the main steam and main feedwater systems to improve plant response to postulated main steam line breaks (MSLB). Subsequently, the licensee reevaluated the benefits that would accrue from these modifications and concluded that the incremental increase in safety was not cost-beneficial. This evaluation discusses the history of the proposed modifications, the staff's review of the licensee's analyses supporting the above conclusion and the staff's judgement concerning the likelihood and potential consequences of a MSLB event at Palisades and whether the proposed modifications are necessary.

2.0 SYSTEM DESCRIPTION

Portions of the main steam, main feedwater and auxiliary feedwater systems at Palisades are depicted in Figure 1. In the steam system, there are two 36" steam lines which penetrate containment each with a main steam isolation valve (reverse check valve) held open by control air pressure. The valves are actuated by containment high pressure or steam generator low pressure signals which vent air from pistons allowing the spring loaded disc to swing into a closed position. Upstream of each main steam isolation valve (MSIV) are 12 safety valves, 2 atmospheric dump valves and the steam supply line to the turbine-driven auxiliary feedwater pump. Downstream of the MSIV is a cross-connect line (26") between the two main steam lines.

There are two main feedwater lines (18"), each line has a regulating valve and stop valve in series. A smaller bypass line (8") around these valves contains a single isolation valve (not shown on Figure 1). The position of the regulating valves is modulated by the three-element (steam flow, feedwater flow and steam generator level) controller.

The original design of the auxiliary feedwater system incorporated one motor-driven pump and one turbine-driven pump. Auxiliary feedwater flow was directed into the main feedwater lines downstream of the feedwater check valves.

During the 1981-1984 time period, several changes were made to the auxiliary feedwater system including addition of a second motor-driven pump, wide-range steam generator level instruments, redundant condensate storage tank level indications, and new auxiliary feedwater discharge piping and spargers. Figure 1 depicts the present system configuration.

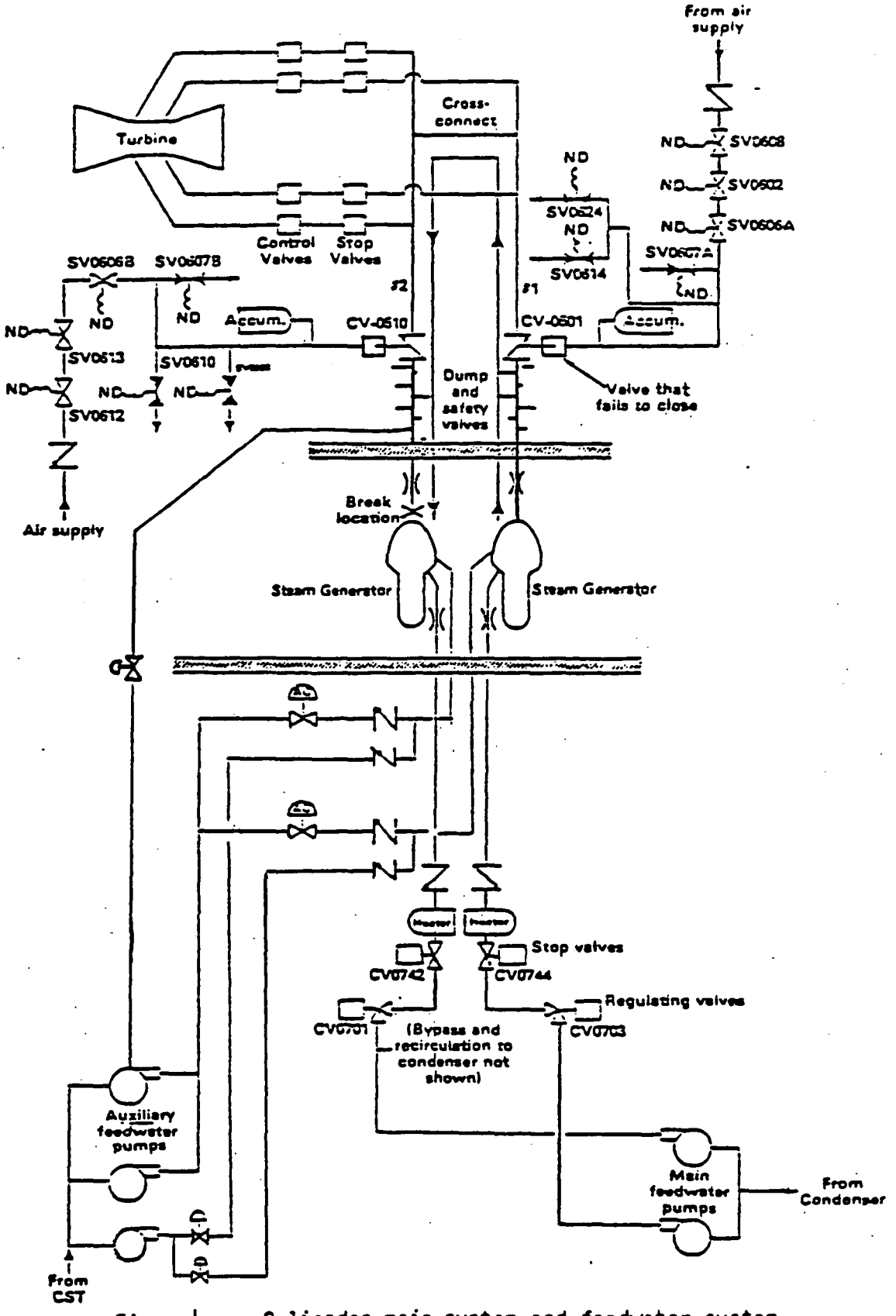

Figure1 Palisades main system and feedwater system

3.0 DESCRIPTION OF OVERALL RESPONSE TO MSLB BREAK EVENTS

This section describes the response of the Palisades Plant to a rupture of high energy piping on the secondary side of the steam generator (i.e., main steam, main feedwater and auxiliary feedwater system piping). The general plant response to failed open atmospheric dump and relief valves would be similar.

For breaks inside containment, steam flow out of the system will result in an increase in containment pressure and temperature. Steam generator pressure and level in the affected generator would drop. The increased heat removal from the reactor coolant system (RCS) causes reactor pressure and temperature to drop. A reactor scram, containment isolation and safety injection actuation signals (SIAS) would be generated.

The MSIV is signalled to close by the containment isolation signal. The high pressure safety injection (HPSI) pumps providing borated water to the RCS are started by the SIAS. If containment pressure is high enough the containment sprays and fan coolers (high-speed) will be started. On low steam generator pressure, the feedwater regulating valve and bypass valve are sent close signals. The AFW system is automatically initiated on low steam generator level.

Safety concerns associated with an MSLB event are as follows:

1) Due to the negative moderator coefficient, the rapid cooldown can result in a reactivity increase, decrease in shutdown margin and recriticality. A return to power could result in localized overheating and fuel damage.

2) the increased containment pressure may challenge the integrity of the containment (for breaks inside containment)

3) the accident environment may affect equipment needed to respond to the event

4) a decay heat removal path must be provided

5) the depressurization load may affect integrity of steam generator tubes

Consideration of the above issues is a factor in establishing design requirements for features of the plant. Furthermore, as discussed above, systems and equipment are included in the plant design to mitigate the effects of a MSLB.

Staff review of MSLB focuses on the areas noted above. In such a review, the staff evaluates the potential for core damage on the basis that no fuel failure is assumed if the minimum departure from nucleate boiling (DNB) ratio remains above a specified value. If the ratio falls below the specified value, rod perforation and release of gap activity is assumed for any rod that fails to meet the criteria. The gap inventory is then used to assess radiological consequences.

The staff also considers the worst possible combination of initial conditions, such as reactor power, temperature, pressure, rod worth, time in core life, consistent with operating limits. The staff postulates a single active component failure of the mitigation systems. In addition, the maximum-worth control rod is assumed to be held in the fully withdrawn position, which reduces the shutdown margin associated with reactor scram and increases the severity in the vicinity of that rod of power peaking.

4.0 BACKGROUND

In late 1978, the NRC initiated the Systematic Evaluation Program (SEP). The purpose of the SEP was to reevaluate the design of eleven of the older operating plants including the Palisades plant, against current licensing criteria to identify areas where the plant design or operation should be improved. As part of this review, the MSLB accident was reassessed. The review criteria used for the staff's evaluation of the MSLB is described in the related Topic Reports of the SEP (ref. Topic Nos. VI-2.D, VI-3, and XV-2). An index of all SEP Topic Report references is located in Appendix E to NUREG-0820, Integrated Plant Safety Assessment, Systematic Evaluation Program, Palisades Plant (October 1982).

The staff noted that for breaks on the reactor side of the MSIV, a single failure of the MSIV in the other loop would allow both steam generators to blow down (See Figure 1). Because of the cross-connect line, only the MSIV can provide isolation between the steam generators for such an event. The MSIV, being a check valve, can only withstand a differential pressure in one direction, namely a higher pressure on the steam generator side of the valve. If the valve in the intact loop fails to close, the steam flow from that generator can flow through the cross-connect line, through the MSIV in the broken loop and out the break.

The staff was concerned that a double generator blowdown could interfere with successful decay heat removal. The loss of steam pressure would cause the turbine-driven AFW pump to be disabled, leaving only one motor-driven AFW pump (the second motor-driven pump had not been installed at the time of the SEP topic evaluation). Also, the staff postulated that the operator may have difficulty controlling this event which is outside the bounds of the original safety analysis. Furthermore, analysis of the containment response resulted in calculated peak pressure of 107 psia, which exceeded the containment design pressure. Because of these uncertainties, the staff identified this issue as important to risk in the integrated plant safety assessment report (IPSAR) for Palisades, NUREG-0820. By letter dated February 16, 1982, the licensee committed to make modifications to make the main steam system single failure proof with respect to this concern.

In a parallel action, IE Bulletin 80-04, Analysis of PWR Main Steam Line Break with Continued Feedwater Addition, identified a concern which could cause the Palisades containment and core response to a MSLB (single steam generator blowdown) to be more severe than the design basis analysis if main feedwater (condensate) flow did not isolate. Following a turbine trip the feedwater control system, acting on the main feedwater pump turbine, reduces feedwater pump flow to 5%; the regulating valve position is not changed. (Note that the turbine-driven main feedwater pumps would trip following MSIV closure which interrupts steam flow to the pump turbine drives.) However, if a steam generator has depressurized and if offsite power is available, the condensate pumps have sufficient pumping capacity to pump more than 5% feedwater flow through the feedwater lines to the steam generator. Therefore, in 1980 the licensee modified the main feedwater system so that the feedwater regulating valve is closed upon low steam generator pressure; the bypass line is similarly isolated. However, the feedwater isolation is not redundant so a single failure could defeat the isolation. By letter dated July 26, 1983, CPCo committed to provide redundant isolation of the main feedwater (isolation signal to stop valve) and bypass lines (valve and signal).

In 1983, the licensee completed a preliminary probabilistic risk assessment (PRA) and cost-benefit evaluation of the proposed MSIV modification. By letter dated August 15, 1983, CPCo requested an extension of the completion date for the MSIV replacement to finalize the evaluation of the safety significance of the MSLB and MSIV failure event and to identify the most cost-beneficial modification (if any). On September 14, 1983, the staff concluded that the extension request was acceptable based on low probability of the pipe break, the successful testing of the MSIV and the expectation that the proposed evaluation of alternatives might result in a more effective corrective action that would provide a broader range of protection (i.e., higher decay heat removal reliability for several events). By letter dated May 23, 1985, CPCo submitted a report including analyses of steam generator and containment response, a PRA, a cost-benefit evaluation and its proposed actions. This assessment addressed both main steam isolation and main feedwater isolation.

5.0 DISCUSSION

The licensee's May 23, 1985 analyses provided the reactor pressure and temperature and steam generator response for both single and double-generator blowdown and for a range of break sizes. Containment pressure and temperature were also calculated. An evaluation was also performed of the effect on steam generator components (e.g., tube sheet) of the blowdown and subsequent reflooding with cold feedwater that could occur for a double-generator blowdown. These analyses were used to justify assumptions and to provide initial conditions for analysis of severe accident sequences associated with a PRA for MSLB and MSIV failures.

The PRA considered as initiating events both breaks in the main steam line piping as well as events which challenged atmospheric dump (ADV) and secondary relief/safety valves. Event trees were then developed considering main steam isolation valve failure, failure to reclose the ADV and relief valve, response of the AFW system, containment heat removal and long-term cooling. In general, a success path occurred if a secondary heat sink through the steam generators could be established in sufficient time to prevent core damage.

A double steam generator blowdown inside containment results in a more severe environment than that for which equipment qualification has been performed. Therefore, the licensee assumed that such equipment would not function and the branches of the event tree resulting in such containment conditions were assumed to lead directly to core melt due to loss of the affected instrumentation.

The specific modifications under consideration in this analysis were the following:

1) Replacement of the existing MSIV with fast acting gate valves. (Other alternatives were considered but were determined to be impractical because of space limitations or turbine control problems).

2) Qualification of instrumentation inside containment for the pressure/temperature conditions resulting from a two steam generator blowdown.

3) Modification of the feedwater stop valves to close on low steam generator pressure.

4) Installation of an additional control valve in series with the feedwater bypass valve which would close on low steam generator pressure.

The licensee's analysis produced the following results:

For MSLB (including ADV/relief valve opening) with MSIV failure, the calculated core-damage frequency is 5.8×10^{-5}/yr. If the single failure potential for the MSIV is fixed, the core damage frequency is approximately 5.5×10^{-5}/yr or a reduction of about 3×10^{-6}/yr. Most of the risk arises from blowdowns outside containment because the initiating event frequencies for valve opening are so much higher than for pipe rupture inside containment. Also, the length of

steam piping upstream of the MSIV, which potentially lead to a double blowdown with one MSIV failure, is a small fraction of the total steam system piping. The related core damage frequency reduction calculated by the licensee for the equipment qualification upgrade is about 10^{-6}/yr.

For the main feedwater stop valve modification, the licensee calculated a core-damage frequency reduction of less than 10^{-7}/yr, and an even smaller value for the feedwater bypass valve installation.

Based on the above results and the costs of the modifications, CPCo concluded that the main steam line isolation valve replacement, upgrade of instrumentation to withstand the double generator blowdown environment and the main feedwater isolation changes are not warranted. The PRA results further indicated that a reduction in core-melt frequency could be obtained if procedures were developed to utilize the condensate system as a backup to the AFW system for decay heat removal.

6.0 EVALUATION

The staff reviewed the licensee's assessment of events such as steam line breaks that would be affected by a failure of an MSIV or main feedwater isolation valve to close. The staff considered the frequency of initiating events, the effects on the plant, the event sequences that could lead to core damage and containment failure, potential offsite consequences, and the benefit that the MSIV replacement, qualification upgrade and redundant feedwater isolation might provide. The results of this review are discussed below.

6.1 Reactor Response

6.1.1 Main Steam Line Break

In the licensing review for Palisades in 1970, for the MSLB accident inside containment, the staff assumed that 10% of the fuel cladding would fail in order to obtain a conservative assessment. The staff calculated 11 rem dose to the thyroid and 1 rem to the whole body at the exclusion boundary, and 3 rem to the thyroid and 0.2 rem whole body at the outer boundary of the low population zone. These doses are well within 10 CFR Part 100 requirements. This was based on consideration of a break at the steam generator nozzle of a main steam line; the other generator was assumed to be isolated by closure of the MSIV. The licensee's analyses for this event predicted that the DNB ratio would drop below 1.3 for less than 1% of the fuel and, thus, cladding is not assured for that portion of fuel. The staff's assumption of 10% failure, therefore, provides an extensive margin to compensate for uncertainities.

In the analyses submitted by CPCo on May 23, 1985, a steam line break size was assumed that maximized the containment pressure for blowdown of one steam generator into containment. This break size was also used for the two steam generator blowdown accident calculations. The original break size selection was based upon the maximum flow rate from the steam generator that would pass steam, and not water, hence maximizing energy transfer. A larger flow rate (break size) would have passed water, according to CPCo, and the total energy transfer from the RCS would have been less. With a two steam generator blowdown, a larger break is possible without passing water and hence, the blowdown can occur more rapidly and RCS cooling would occur over a shorter time, which would influence DNB response. The staff believes that the RCS short-term pressure response would be less sensitive to the blowdown conditions than the RCS temperature due to upper head voiding considerations. Therefore, the fuel rod temperatures will be the most important result to address.

Clearly, the transient response of the plant may be different for the two steam generator blowdown than that analyzed by CPCo. More rods may experience a short-term DNB condition or they may attain a higher temperature than would occur for a single steam generator blowdown. However, the staff believes that DNB would be restricted to the vicinity of the assumed stuck control rod (some few fuel bundles) and that fuel rod temperatures would at worst be within a few hundred degrees (perhaps 200°F) of previously predicted values. Examination of the 10% assumed fuel clad failure (by the staff),

and consideration of the transient power distribution in the vicinity of the postulated stuck rod would lead one to conclude that the 10% failed clad assumption is probably still applicable (and bounding), but plant-specific analyses have not been done to confirm this judgement. Based on this judgement and the previous dose results, the Part 100 siting guidelines are likely still satisfied for this event.

Analysis of the MSLB event at Palisades shows that the continuous addition of feedwater, which could result from a single failure of a feedwater isolation valve as discussed previously, does not worsen the reactor power transient. This transient occurs so rapidly after the MSLB that there is not enough additional water put into the steam generator in that time to make any difference in the amount of fuel failures.

The staff also considered the capability to establish a decay heat removal path following a MSLB with failure to isolate the intact steam generator, because secondary side pressure could not be directly controlled. In turn, it may be difficult to control secondary side heat removal and, consequently, the primary cooldown rate. Although the system response for a two steam generator blowdown is not substantially different, the staff concludes that procedures based on systems analyses should be developed (see Section 6.5) to enable the operator to cope with the event and to achieve a controlled cooldown as soon as possible after the initial steam generator blow down.

6.1.2 Steam Generator Tube Rupture

The plant response to a steam generator tube rupture would also be affected by MSIV failure. A single failure of one of the two, which would allow back flow, could prevent the isolation of a steam generator with a ruptured tube. If the MSIV on the steam generator that has the ruptured tube fails to close, steam that is contaminated with primary reactor coolant could flow through the failed MSIV, through the cross over pipe, back through the other MSIV, and out the atmospheric dump valves (ADV). Thus, primary coolant would be continually released to the atmosphere through the ADV's until the plant is cooled down to conditions where the Shutdown Cooling System (SCS) could be used.

CPCo has analyzed a case that is equivalent to this and has discussed it in the Palisades Final Safety Analysis Report (FSAR), Section 14.15.4. A steam generator tube rupture is assumed with a loss of offsite power and no isolation until the reactor coolant system is cooled to 325°F (SCS initiation temperature). The offsite radiation doses due to this scenario are less than the guidelines of 10 CFR Part 100.

6.2 Containment Response

During the SEP review of Palisades, the response of the containment was evaluated for a variety of accidents; the results are presented in a staff evaluation transmitted by letter dated November 16, 1981. It was concluded in that evaluation that containment conditions could be as high as 107 psia and 420°F (if offsite power is available) or 98 psia and 465°F (if offsite power is lost) if both steam generators blow down through a ruptured steam line to containment.

For the case of continued feedwater addition with one steam generator blowdown, the staff considered the consequences on containment in comparison to the two steam generator blowdown. Analyses of the two steam generator blowdown at Palisades show that the containment pressure peaks about 2.5 minutes after the MSLB occurs. The magnitude of this peak is high because it takes the containment sprays and air coolers about 1 minute to attain design cooling capacity.

The containment pressure with only one steam generator blowing down cannot rise as rapidly even though it has continuous feedwater flow. In this case, the containment sprays and air coolers would be more effective. This would result in a lower peak containment pressure for the case with continuous feedwater addition than for the case with a failed MSIV and a two steam generator blowdown.

The effects of containment pressure and temperature on the containment for the bounding condition of the two steam generator blowdown are discussed below.

6.2.1 Containment Structural Integrity

The Palisades Plant uses a dry prestressed concrete steel lined containment structure with a free volume of 1.64 million cubic feet. The design pressure is 55 psig or 70 psia. Based on the information presented in the FSAR, the containment structure was analyzed for the combination of dead load plus prestressing load plus 1.5 times the design pressure (1.5 X 70 = 105 psia) plus the thermal loads due to the design basis accident. For this load combination the load factor approach was used to assure a low strain elastic response. The minimum margin was in the prestressing tendons which is at least 5% above the combined loads considered. Under the postulated MSLB situation, the worst predicted internal pressure is 107 psia which is about 2% above 1.5 times the design pressure (105 psia). Considering the margin available above the combined loads analyzed in the FSAR, the containment response for the postulated new loading should be essentially elastic. It should be noted here that the higher peak temperature in the postulated loading lasts for only a few minutes and is judged not to be detrimental to the structural integrity of the containment since the thermal inertia of the containment structure is quite substantial. Therefore, the staff concludes that the integrity of the containment structure would not be seriously challenged by the pressure from this postulated event.

6.2.2 Containment Leakage

The staff has reviewed the effect of the higher temperature on the seals and gaskets for containment closures. All the containment penetration closures are a pressure-seating type with seals protected from exposure to the direct containment atomsphere by metal flanges. Recent tests on seals and gaskets performed by the Sandia Laboratories for the Office of Research indicate that seals and gaskets are not likely to fail under the postulated containment conditions.

Based on the staff review performed under the SEP, it has been determined that all the pipes penetrating containment have isolation valves outside containment. Therefore, a malfunction of any of these isolation valves as a result of higher containment internal conditions following the postulated accident condition is not likely.

6.2.3 Equipment Qualification

Equipment relied upon to remain functional following design basis accidents should be qualified for the environment it may experience as a result of the accident. In its review of environmental qualification of electrical equipment for Palisades, the staff used its "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines). Both loss of coolant accidents (LOCA) and MSLB accidents are considered.

Equipment qualified to the DOR Guidelines for a LOCA environment is generally considered qualified for a MSLB accident environment in plants with automatic spray systems not subject to disabling single component failures. This position is based on the "Best Estimate" calculation of a typical plant peak temperature and pressure and a thermal analysis of typical components inside containment.

In the case of Palisades, the environmental conditions resulting from a two-steam-generator blowdown are not typical. The resulting peak temperature of 465°F shown in Figure 4.7T of the November 16, 1981 evaluation would, in all likelihood, result in certain equipment inside containment achieving temperatures above LOCA qualification temperatures. Additionally, the pressure shown in Figure 4.6P or 4.7P of the same report could cause certain equipment to fail from moisture intrusion. These conclusions are based on the staff's experience with near-term operating license and operating reactor reviews of equipment qualification for MSLB and of equipment survivability in a hydrogen burn environment. The peak pressure and temperature resulting from the two-steam-generator blowdown are significantly higher than environmental conditions to which most equipment has been qualified.

As stated above, Palisades is equipped with an automatic containment spray system not subject to disabling single failures. However, the temperature for the MSLB inside containment exceeds the LOCA profile (278°F) by almost 200°F, albeit for a short time period (approximately 2 minutes). In view of the above considerations, the staff therefore agrees with the CPCo assessment that electrical equipment inside containment may not remain functional during and following the two-steam-generator blowdown event. Section 6.5 of this evaluation discusses compensatory measures needed to account for possible failure of this equipment.

6.3 Steam Generator Integrity

The analyses associated with the structural integrity of the CPCo steam generators at Palisades are contained in the Combustion Engineering (CE) Report CENC-1120 "Analytical Report for Consumers Power Steam Generator." The design analyses contained therein are based upon original pressure and temperature parameters which are less severe than the two steam generator blowdown due to reduced differential pressures on the tubesheet

and the steam generator tubes. Staff review of the design analyses contained in this report indicates that steam generator vessel, tubesheet, or tube failure is not likely as a result of steam generator blowdown which results in a dryout during which the primary side is at 2500 psia and 600°F and the steam generator secondary side subsequently refloods with cold feedwater.

The steam generators are designed such that no component will fail by rupture or by developing deformations (elastic or plastic) that impair the function, performance, or integrity of the steam generator when subjected to, among other transients, eight cycles during which the primary side is at 2500 psia and 600°F while the secondary side is at atmospheric pressure and eight cycles of adding a maximum of 300 gpm of 70°F feedwater with the steam generator dry at 600°F.

In the analyses provided in CENC-1120 the licensee has made conservative estimates of thermal hydraulic loadings which bound the loadings that may occur during unanticipated non-operating transients. The tubesheet is considered as most vulnerable to temperature differentials because it is thick and because it is confined by the stay on the inside and the outer steam generator shell on the outside. This thickness permits the development of large temperature differentials and ensures a large difference in the propensity for thermal growth between the top and bottom surfaces. Even large temperature differences, however, do not imply large stresses in the tubesheet unless the free end displacement of the tubesheet is constrained, as it is by the stay and shell.

An investigation of the vulnerability of the steam generators to large temperature differentials has focused on the tubesheet and its shell/stay connections. Limiting stresses have been found where the tubesheet and stay meet or where the tubesheet and shell meet.

The CE analysis meets the ASME Code, Section III NB-3200 requirements. Basic ingredients of the analysis are as follows:

- evaluation of the primary (essentially pressure or weight induced) loads to ASME Code limits to provide protection against catastrophic failure.

- comparison of primary plus secondary (induced by thermal growth) stress range with an acceptance criterion of 2 Sy to ensure that after a few cycles, components will "shakedown" to elastic response and that plastic ratchet action will not take place after a few cycles,

- comparison of applied loading cycles to allowable loading cycles to ensure that the fatigue usage factor is less than 1.0 (this implies protection against fatigue-cracks),

- utilization of Appendix A of the ASME Code, Section III which enables one to analyze the perforated plate of known hole pitch and ligament within terms of an "equivalent solid plate," and

- assumption that the film temperature drop at the secondary side of the tubesheet was zero.

The licensee demonstrated by analysis in its repair report of April 19, 1984, that plugging of tubes with cracks or with intergranular attack (IGA) defects of greater than or equal to 51% through-wall depth was sufficient to meet all safety and regulatory criteria including the requirements of Regulatory Guide 1.121. The licensee demonstrated that defects equal to or less than this magnitude would be able to withstand, without failure, the blowdown loads of MSLB. This plugging limit also takes into account the uncertainties associated with eddy current measurements and probable allowance for degradation until the next inspection. Leak-before-break was demonstrated for certain through-wall cracks of limited extent such as cracks and IGA which had been observed at Palisades steam generators. In the event of a MSLB, these defects are expected to result in a leak rather than a tube rupture or severance.

The steam generator tubes in a CE steam generator are free to expand or contract axially unlike those in a Once Through Steam Generator (OTSG). Cold feedwater impingement on hot tubes during a dryout condition can result in a thermal mismatch between the shell and the tubes prior to attainment of steady state conditions. This thermal mismatch can result in axial compressive loads on the tubes in an OTSG since the tubes are constrained between the upper and lower tube sheets. However, in a CE steam generator no such axial loads would be produced. Thus, the thermal loadings on the tubes themselves is not a concern during cold feedwater injection at the Palisades steam generators. There is also no effect on the fatigue usage factors, which had been shown to have a large margin in the original design.

The CE analysis is judged to be conservative. In the tubesheet analysis, CE tends to lump secondary stresses into the primary load category, uses skin stresses in meeting the primary plus secondary stress intensity range limit and employs no film drops. These assumptions result in a somewhat more conservative analysis than that required by the ASME Code. Appendix A of the ASME Code has been followed rigorously. The result is that all ASME Code requirements are met.

The staff, therefore, concludes that steam generator vessel, tubesheet, or tube failure is not likely as a result of steam generator blowdown resulting in dryout and subsequent reflooding with cold feedwater.

6.4 Risk Assessment

The licensee provided a detailed PRA in the May 23, 1985 submittal to examine the impact of the issues of redundant main steam isolation, equipment qualification and redundant main feedwater isolation, on the potential risk from secondary system failures. This study included detailed event trees and fault trees associated with the systems of interest and analyses to estimate offsite consequences.

The staff's review of this evaluation has considered the core melt frequency from these steam system failures, and the offsite consequences that would result from combinations of core melt and containment conditions (such as early core melt, with containment bypass, with containment integrity or with late containment failure). For the cost-benefit evaluation, the calculated person-rem release for an event is multiplied by the reduction in frequency of occurrence of that event associated with the modification and by the number of remaining years of plant operation to obtain the averted risk. For example, if a modification can result in a 10^{-5}/reactor year (RY) reduction in frequency of an event, the person-rem release for this event is 10^{6} rem, and there are 25 years of plant life remaining, the potential averted risk is 250 person-rem ($10^{-5} \times 10^{6} \times 25$). Using $1000/person-rem, a cost of $250,000 to implement such a modification would tentatively be considered justifiable on the basis of averted risk.

6.4.1 Evaluation of Core Melt Frequencies

Accident sequences to be considered in the evaluation of each issue (i.e., steam line isolation; equipment qualification; feedwater isolation) depend on the unique characteristics associated with that issue. It is not necessary to assess all accident sequences, only those that are directly impacted. For instance, failure of long-term heat removal may be independent of whether one or two steam generators blow down following a steamline break. Therefore, it may not be necessary to evaluate long-term cooling scenarios with respect to the impact of blowing down one versus two steam generators.

In some instances, the licensee has examined a larger set of accident sequences than is necessary; however, certain events of potential interest were not included. Therefore, the staff's evaluation of these issues was a reformulation of the risk estimates using pertinent information from the licensee's study along with findings from other related work.

6.4.1.1 Redundant Steam Line Isolation

As discussed previously, if a particular MSIV does not close following secondary system accidents, the operator will not be able to isolate a faulted steam generator (line) from the good steam generator (line). Thus, certain secondary system accidents could be aggravated by the single failure of an MSIV to isolate. The types of core-melt scenarios of interest to steamline isolation are those associated with loss of secondary heat removal capability, RCS overcooling that could result in pressurized thermal shock, and loss of RCS inventory outside of containment.

The following evaluations cover the types of situations that are pertinent to the consideration of redundant steamline isolation:

6.4.1.1.1 Loss of Secondary Heat Removal Capability

Following an accident, decay heat removal is normally accomplished by the three-train AFW system which has two motor-driven pumps and a steam-driven pump that discharge into the two steam generators through two sets of lines and associated valving. If both steam generators blow down following a

steamline or feedwater line break, then only the two motor-driven AFW pumps would be available for decay heat removal instead of three AFW pumps. Thus, the incorporation of redundant main steam line isolation would increase the probability of the availability of AFW following a break in the secondary system upstream of the MSIVs.

The two breaks of interest, those inside containment and those just outside of containment, upstream of the MSIVs, were analyzed by the licensee. For breaks inside containment, the licensee assumed that the instrumentation (namely steam generator level) needed for long term heat removal would fail in a two-generator blowdown environment, which is more severe than the instrumentation design envelope. Thus, core-melt was postulated to occur if the MSIV failed to isolate the "good" steam generator following a large secondary break inside containment. Thus, the core-melt frequency is assumed to be equal to the probability of a large steam line break times the probability of MSIV failure.

The licensee estimated a core-melt frequency of 10^{-6}/RY for this sequence. The staff would estimate a frequency of 2 to $3x10^{-6}$/RY because of a somewhat higher secondary line break estimate. Staff review of steam line breaks for the pressurized thermal shock studies (SECY 82-465) would result in estimates of break frequencies of about 10^{-4}/RY for two loop plants compared to the licensee's value of $4x10^{-5}$/RY.

The core-melt frequency attributed to this sequence can probably be reduced by 90% by implementation of redundant steam line isolation. Thus, the potential benefit is a reduction in core-melt frequency of $2x10^{-6}$/RY. This benefit may be somewhat overestimated because of the assumption that all critical instrumentation fails in a two-steam generator blowdown and because of the failure probability of the MSIVs calculated by the licensee. In the early 1970s, there were several failures of the MSIV at Palisades; however, there have not been any recent failures of the MSIVs (none in the last 12 years) so that simple averaging of the early failures over the elapsed plant operating time may be conservative.

The second steam line break of interest is in the piping outside of containment and upstream of the MSIVs. These sequences include stuck open steam generator safety and relief valves. Under those conditions, the blowdown of two steam generators (because of a failure in the MSIV on the "good" steam generator) would defeat the steam-driven AFW pump. Consequently, core melt would occur if the piping fails (or a valve sticks), the MSIV fails and the two-train system fails.

The licensee has estimated the frequency of a stuck-open steam generator relief/dump valve to be about 10^{-1}/RY based on operating experience. Palisades experienced stuck open relief valves early in plant life. It has also estimated the failure probability of the MSIV to be about $2x10^{-2}$ based on the operating experience discussed above. Based on the licensee's fault trees, assuming recovery from certain common cause failures, and assuming the operator will establish a secondary heat sink even if two steam generators blow down, the staff estimated the unavailability of the two-pump portion of the AFW system to be about $2x10^{-3}$. Combining the above basic events results in a core-melt estimate of $4x10^{-6}$/RY.

The staff believes that the frequency of this sequence can be reduced by at least 90% by implementation of redundant steam line isolation yielding a reduction of about $3x10^{-6}$/RY (about the same value estimated by the licensee). The dominant assumption in this analysis is that the operator would use one of the faulted steam generators to remove decay heat, thus leaving hardware failures as dominant contributors to loss of AFW.

A cursory review of the emergency procedures related to secondary line breaks (EOP-6 and EOP-7) and normal reactor trip (EOP-1) leaves the impression that the operator is not given clear instructions to ensure he will respond properly. The human error associated with improper heat removal could dominate this sequence and make it higher by 10 to 100; this concern is further addressed in Section 6.5.

Another bias in this analysis is that it does not include any allowance for potential feed and bleed to remove decay heat or for the utilization of the condensate system to provide feedwater. The licensee has proposed to develop the appropriate procedures for the condensate system as an alternative to providing redundant steam line isolation. These considerations would reduce the estimated benefits of proposed modifications.

Combining the two estimates, the potential core melt reduction is about $5x10^{-6}$/RY for these events.

6.4.1.1.2 Pressurized Thermal Shock

The licensee has addressed the issue of pressurized thermal shock associated with the blowdown of two steam generators following a steam line break or equivalent and the continuation of AFW. The staff concurs that the pressurized thermal shock screening criteria appears to envelop this type of event. The staff bases this conclusion on a comparison of this overcooling event (stuck open safety/relief valve initiator) frequency for Palisades ($2x10^{-3}$/RY) with a staff evaluation for B&W plants in SECY 83-288 which had an event-initiating frequency of about 10^{-2}/RY. Even if the operator fails to isolate the stuck-open dump valve, the staff judges that the resulting temperature-frequency distribution would still fall within the envelope considered in the pressurized thermal shock screening criteria (SECY 82-465). It is noted, however, that emergency procedure EOP-7 (secondary system line breaks) does not contain instruction regarding isolation of stuck open secondary valves or any consideration of coping with overcooling situations. It may be prudent to reexamine the operator actions for this type of event since the estimated initiating frequency is about 2×10^{-3}/RY. (See Section 6.5).

6.4.1.1.3 Steam Generator Tube Rupture (SGTR)

The normal operator response to an SGTR is to isolate the faulted steam generator and cool down the RCS with the good steam generator until the residual heat removal system can be activated. If the MSIV fails to close on the faulted steam generator, there will be continuous loss of primary inventory from the faulted steam generator through the good steam generator to the atmosphere or to the condenser. The concern examined by the staff is whether the operator can terminate the continuous loss of reactor coolant prior to exhausting the safety injection and refueling water tank (about 250,000 gallons).

From NUREG-1044, the estimated frequency of one tube rupturing is 10^{-2}/RY; for two tubes it is $2x10^{-3}$/RY; and for ten or more tubes it is $2x10^{-4}$/RY. From NUREG-0844, the estimate of human error probability of not recovering from a single tube rupture is 10^{-3}/demand(D); for two to ten tubes it is 10^{-2}/D; and for more than ten tubes it is 0.1/D. Combining these probabilities with the estimated failure probability of $2x10^{-2}$/D for the MSIV results in a core-melt frequency of about 10^{-6}/RY.

This core-melt estimate may be somewhat conservative because the coolant loss would be terminated once residual heat removal cooling was established if the steam generator safety valve is not stuck open. The emergency procedure EOP-8.2, for steam generator tube rupture, calls for rapid plant cooldown even using the faulted steam generator; however, there is an absence of guidance on the potential problems of blowing down both steam generators with the attendant loss of reactor coolant inventory. This concern is also addressed in Section 6.5.

6.4.1.1.4 Summary for Main Steam Isolation

Implementation of redundant isolation of the main steam lines would result in an estimated reduction in core-melt frequency of about $6x10^{-6}$/RY. This estimated reduction in core-melt frequency is smaller than the 10^{-5}/RY objective cited in the ATWS rulemaking (SECY 83-293) and the station blackout rulemaking (SECY 85-163A).

6.4.1.2 Modification of Envelope for Equipment Environmental Qualification

The blowdown of two steam generators inside of containment would result in conditions beyond the design qualification for critical equipment inside containment. Modifying the environmental qualification profile to cover the conditions resulting from the two generator blowdown would provide added assurance of the operability of selected equipment inside containment. As noted in Section 6.4.1.1, the staff estimates the frequency of a main steamline break inside containment and subsequent failure of the appropriate MSIV to be 2 to $3x10^{-6}$/RY. If the equipment inside containment such as steam generator level indication (used to control removal of decay heat through the steam generators) is qualified to more severe conditions, the staff estimates that the core-melt frequency for this event could be reduced by 90%, or a reduction of about $2 X 10^{-6}$/RY.

6.4.1.3 Redundant Feedwater Line Isolation

Palisades has two main feedwater lines, each with a regulating and stop valve. There are smaller bypass lines around each regulating and stop valve that have one valve each that perform both control and isolation. The bypass lines are only open during startup and shutdown evolutions when the main feedwater lines are isolated.

Only the regulating valves receive a close signal on low steam generator pressure or high steam generator level. The steam admission valves to the turbine drive are ramped down following a reactor trip; however, the condensate pumps still operate and could inject water into a depressurized steam generator even with an idle main feedwater pump if a regulating valve fails to isolate.

The licensee examined the lack of redundant feedwater isolation from the standpoint of loss of decay heat removal following a main steam line break and determined that it was a small core-melt contributor ($< 10^{-6}$/RY). The staff concurs in this determination; however, the staff also examined potential overcooling (leading to pressurized thermal shock) caused by excessive feedwater flow following a reactor trip.

There are two overcooling sequences of interest: a plant trip with continued main feedwater and a small steam line break (stuck open ADV) with continued main feedwater. Based on discussions with the licensee there are 469,000 pounds of water in the RCS at 560°F and only 500,000 pounds of water in the hotwell at 70°F. The average temperature resulting from adiabatic transfer of heat to the hotwell water is above 300°F. This estimate ignores sensible heat in the core, structure, and steam generator water and the slow heat losses from a small steam line break. On balance, the staff believes that the resulting fluid temperatures in the pressure vessel will remain above the range of interest to pressurized thermal shock (about 200°F).

6.4.2 Evaluation of Containment Failure Modes

This section of the evaluation discusses the interactions of core melt with the primary containment structure. Some of the failure modes discussed below are applicable to any core melt scenario not just for those initiated by secondary system failures.

6.4.2.1 Vessel Cavity/Containment Sump Configuration

Palisades has a large containment volume and the high pressure required to cause failure (209 psi ultimate as calculated by CPCo) places it in the category of other pressurized water reactors with large dry containments. The emergency safety injection system sump is unique; it is located under the reactor vessel cavity, as shown schematically in Figure 2. The sump is about 3.5 feet high, 22 feet in diameter, and also contains a smaller sump for control of normal containment drainage.

There are five major connections between the lower vessel cavity and the remainder of containment:

1. A flanged-closed access passage

2. An annular space between the reactor vessel and the reactor vessel cavity

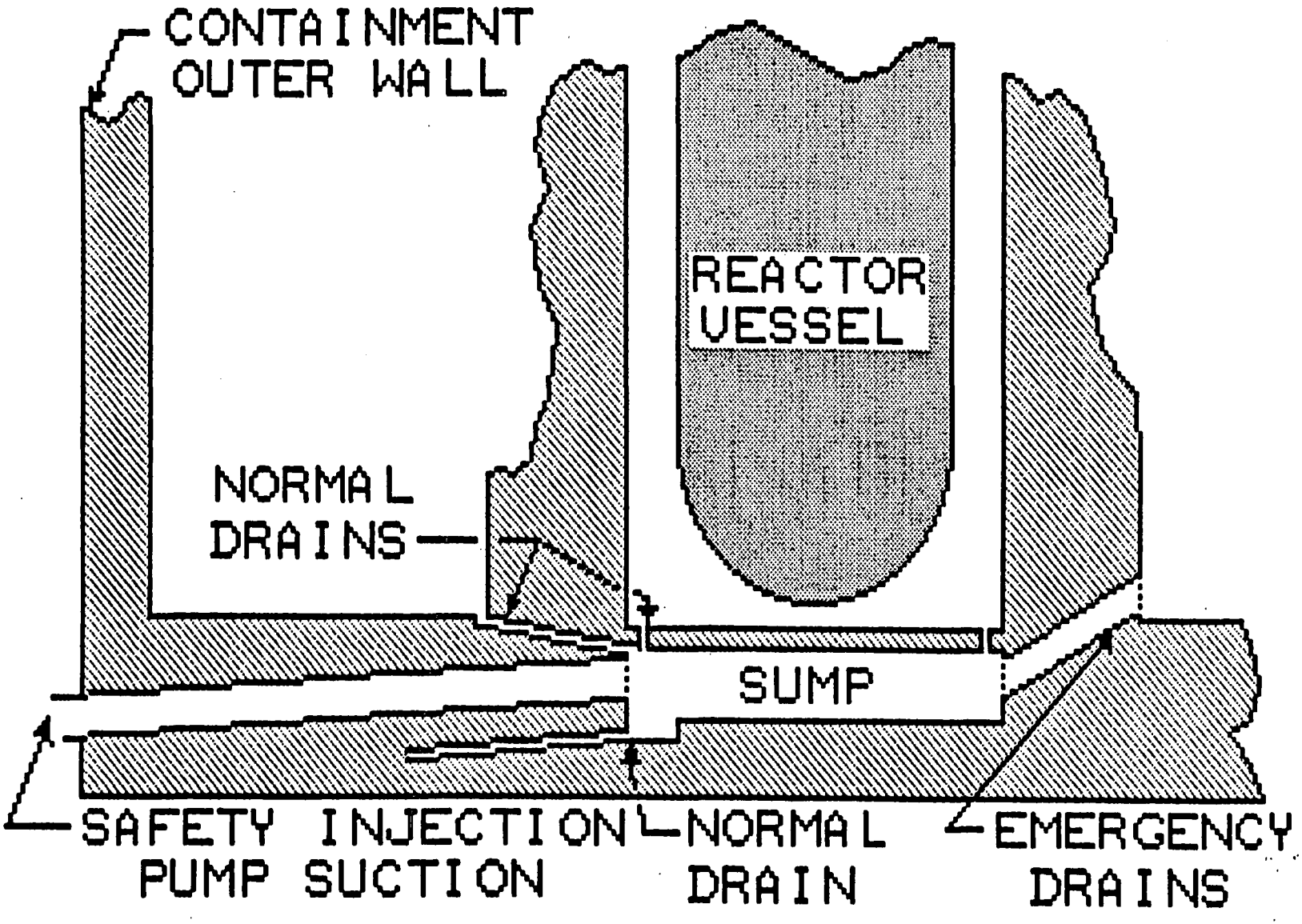

FIGURE 2. PALISADES SUMP

3. Space between the reactor vessel piping and the reactor vessel cavity wall

4. Drain connections from containment

5. Two 1-inch drains from the reactor vessel cavity floor to the sump

6.4.2.2 Influence of Configuration During Severe Accidents

The likelihood of a breach in the RCS at a location other than the lower head of the reactor vessel is uncertain. Therefore, the staff cannot exclude high pressure failure of the lower head, and it is necessary to address possible chemical reactions between melt material and the containment atmosphere at the time of RCS pressure boundary failure.

If a high pressure blowdown of melt were to occur through the reactor vessel bottom, the first two items in the above list would probably provide the largest vent paths. The others are probably too small to affect short-term containment response. Melt ejected from the reactor cavity via the access passage would encounter a tortuous path prior to reaching upper containment. The staff opinion is that most melt entering this path would be removed before reaching the large open volume of upper containment. Melt ejected up the annular space would be restricted at the seal region, and would impact upon the missile shield prior to reaching upper containment. The staff judgement is that most of the melt entering this path would not pass into upper containment as an aerosol, which would be necessary for a significant chemical reaction to occur.

The staff concludes that most of the melt will not reach upper containment in a form suitable for reaction with the containment atmosphere, and a chemical reaction of sufficient magnitude to fail containment is unlikely. However, the staff cannot exclude the possibility that a large quantity of molten core material can react with the upper containment atmosphere in of the order of milliseconds, thereby producing a pressure pulse which ruptures containment. Therefore, a small conditional probability was assigned to containment rupture given a blowdown of melt material through the bottom of the reactor vessel with the reactor vessel at high pressure.

The dynamic interaction between high pressure melt ejected through the bottom of the reactor vessel and the concrete floor immediately below has not been analyzed, but the staff judges that the floor would not survive. Even if short-term survival did result, drains between the bottom of the reactor vessel cavity and the sump would provide a path for passage of melt and the flowing melt material would probably increase the diameter of the drains. This path also would be applicable to melt-through of the lower vessel head with the RCS at low pressure. Therefore, the conditional probability of a significant quantity of debris being deposited in the sump at the time of reactor vessel breach is considered to be one.

Debris in the sump may block the pipes leading to the safety systems, or debris may be sucked into the pumps. The staff will assume the conditional probability of one that significant blockage of these pipes will occur, given the presence of a significant quantity of melt material or debris in the sump. It further appears reasonable to assume that any pumps using the sump as a source of water will be destroyed or badly damaged if they are in operation at the time of reactor vessel breach. The conditional probability of this reaction will also be assumed to be one.

Seal damage which releases a significant quantity of radioactive material outside of containment may be a result of pump operation during the above discussed conditions. Movement of melt into the pipes, which are 24 inches in diameter and have a slight slope toward the outside of containment, could lead to penetration of the containment boundary with release into the auxiliary building. Other failure mechanisms can be postulated which would fail the pipes, including short-term overpressure in response to molten material interacting with water in the vicinity, and blockage of the entrance region of the pipes followed by pressurization from heating of trapped water. None of these has been investigated. For purposes of this evaluation, the staff will assume a small conditional probability for failure of suction lines leading to bypass of the containment boundary given the release of molten core material into the sump.

6.4.3 Evaluation of Offsite Release Calculations

A realistic assessment of containment response to a core-melt accident is a complex task requiring plant-and site-specific analyses. This work has been accomplished for several plants and sites with containment designs similar to Palisades. The staff has assumed the results can be applied to Palisades. The uncertainty in this approach is high, but significant insight to expected plant response can be obtained.

Conditional offsite consequences of core-melt accidents generally depend on whether the containment fails early, late, or not at all; and on whether containment sprays operate to reduce fission product inventory prior to failure. The most severe case in terms of radiological consequences is early failure with no sprays. If containment failure is delayed for several hours following reactor vessel melt-through, fewer radionuclide aerosols are released and offsite consequences are significantly reduced. Even lower consequences result if the fission product suspended aerosol inventory is reduced by containment spray operation before or during containment failure. However at Palisades, due to the sump configuration, the staff assumes failure of containment sprays in the recirculation mode, given a core-melt event. The licensee has made the same assumption. If the containment does not fail, or fails by basemat melt through, offsite consequences are low enough to be negligible with respect to most cost/benefit considerations.

Containment response is a function of the accident and the available mitigation equipment. A steam line break inside containment would normally actuate fan coolers and sprays early in the accident. This will rapidly reduce pressure and temperature. Additional failures are necessary to achieve core melt. Following the double steam generator blowdown and failure of all RCS decay heat removal mechanisms there would be roughly an hour before core melt. This is sufficient time for fan coolers and sprays to have returned containment conditons to close to the pre-accident condition. Fan coolers and sprays would be actuated by a high containment pressure signal. The corresponding transducers are physically outside containment and not subject to the harsh environment. The staff considers it almost certain that core melt under these containment conditions will not cause early containment failure unless containment is bypassed. This conclusion is based on the findings of a number of PRAs, and includes consideration of such effects as early hydrogen burns (probable, but of insufficient magnitude to jeopardize containment), steam explosions (of the order of 10^{-4} likelihood or less), so-called direct heating (of insufficient magnitude to affect containment as previously discussed), and the pressure pulse associated with blowdown of the RCS from approximately 2400 psi at the time of lower vessel head failure.

Based on experience with this class of containments, it is likely that containment integrity will be maintained for about 12 hours to a day or more if containment cooling is lost at accident initiation. Cooling failure later in the accident would typically cause containment failure days later, if at all. The high failure pressure associated with the Palisades design would lead one to expect a longer time in comparison to plants with lower pressure capability containments.

As previously discussed, the staff considers it unlikely that containment spray will be available following core melt and depletion of the emergency water supply unless means are taken to replenish the water in the emergency water storage tank. Satisfactory operation of fan coolers is questionable due to clogging with debris and aerosols and they are not qualified for the conditions that may exist for a short time following blowdown of two steam generators inside containment, but they may work for a time. If fan coolers or sprays can be operated for several hours following lower reactor vessel head failure, the staff considers it likely that the Palisades containment will remain intact for several days following a core-melt scenario, and may not breach in the long term. If containment spray or fan coolers (assuming a significant water inventory in containment) can be operated on an intermittent basis in the long term, the staff considers it unlikely that containment failure will occur at all.

Late containment failure due to hydrogen burns is unlikely due in part to the high likelihood of early burns which deplete both hydrogen and oxygen and limit the reactants available for later burns, and due in part to the strength of the Palisades containment.

Bypass sequences may be summarized as follows:

1. Early failure due to consequential failure of the 24-inch pipes leading from the emergency sump.

2. Early bypass due to an open ADV in conjunction with the steam line break.

3. Failure to isolate containment.

4. Containment basemat melt-through.

5. SGTR caused by the response to the steam line rupture.

6. SGTR caused by heating of the tubes from core heat.

Early failure of the lines leading from the emergency sump and containment basemat melt-through have been discussed previously. Item 3 was not addressed in the CPCo analysis the staff reviewed. The staff included this mode of containment bypass in its independent evaluation.

Item 6 was not addressed in the CPCo evaluation of the Palisades steam line break. The staff does not believe this condition will occur, but cannot exclude it. This bypass accident is assigned a low conditional probability by the staff and is included in the staff evaluation.

CPCo identified a unique combination of circumstances at Palisades which could result in containment bypass during a steam line break inside containment. The controls of the ADV on the steam generator secondary side are set to open the valves if the average reactor coolant temperature becomes too high. This will occur if RCS heat removal is lost or during an approach to core-melt conditions. This action will open the steam generator secondary side to the environment. Any steam line break inside containment at Palisades will open a path between containment and the steam generator secondary system. Consequently, a flow path will exist for venting the containment atmosphere to the environment via the steam line break and the ADV. CPCo assumed a 0.995 probability that the operators would close the valves, despite the lack of any procedures which cover the situation, and despite the more normal condition that one would want the valves open to enhance steam generator secondary side cooling. The staff considers the CPCo assumptions to be unrealistically optimistic; this issue is further discussed in Section 6.5.

The final containment bypass sequence involves SGTR. This is also a situation in which the actions of the ADV influence releases by opening a direct path to the environment. The staff has considered the various combinations of failures which may occur, and concludes that the overall effect is an increased likelihood of occurrence of a SGTR which affects both steam generators. However, there is little impact upon risk in comparison to other bypass conditions which may exist under severe accident conditions.

For medium-population density sites such as Palisades, early containment failure would result in an offsite dose of about $2x10^7$ person-rem. Several orders of magnitude reduction in dose would result if containment failure were delayed, depending on the duration of the delay. A value of $2x10^5$ person-rem is assumed for such events. For basemat failures or cases with no containment failure, the consequences are estimated to be no greater than $2x10^4$ person-rem. The staff believes that $2x10^6$ person-rem is a cautious weighted average of potential consequences.

Several accident conditions were selected and dose estimates obtained by the staff for those conditions. The results are summarized in Table 1. The weighted consequences (person-rem) obtained from the staff's review were then combined with the core melt frequency reductions discussed in Section 6.4.1 for the cost-benefit evaluations in the following section.

6.4.4 Cost-Benefit Evaluation

Using the staff's estimates of core-melt frequency, releases and 23 more years of plant operation, the staff estimated the benefit (justifiable cost) of proposed modifications. These estimates are based on offsite consequences within 500 miles. Major population centers east of the site are Detroit (175 miles) and Lansing (100 miles). Chicago and Gary are located about 80 miles south-west of the site. The benefit would be reduced by up to a factor of 2 if only a 50-mile radius is considered.

For the mainsteam line isolation, the averted offsite consequences are worth $310,000. The actual cost of the MSIV replacement is estimated by the licensee to be in excess of two million dollars. For the equipment qualification envelope upgrade of instrumentation inside containment, the associated justifiable cost would be about $40,000 based on the assumptions noted in Section 6.4.1. This estimate is conservative because of the assumptions that all instrumentation is lost and that the operator would not maintain decay heat removal. The licensee estimates the cost of this upgrade to be one million dollars. For the redundant feedwater isolation case, the staff concludes that averted offsite consequences are worth about $20,000 (justifiable cost). The licensee estimates the cost of modifying the actuation logic for the feedwater stop valve to be about $30,000, and for provision of redundant isolation of the bypass line about $200,000.

The licensee has also noted a potential decrement in plant safety associated with the feedwater isolation modifications for situations where the AFW system is not functioning for decay heat removal and use of the condensate/feedwater lines would be beneficial; the redundant feedwater isolation reduces the reliability of the condensate system.

TABLE 1. DOSE ESTIMATES FOR SELECTED SEVERE ACCIDENT CONDITIONS

TYPE NUMBER *	ITEM	CONDITIONAL CONSEQUENCES PERSON REM	% OCCURRENCES PER CORE MELT	WEIGHTED CONDITIONAL CONSEQUENCES PERSON REM
1	STEAM GENERATOR TUBE RUPTURE	1×10^7	1	1×10^5
	BYPASS VIA CONTAINMENT SUMP	4×10^7	1	4×10^5
	FAILURE TO ISOLATE	1×10^6	0.5	-
	EARLY CONTAINMENT FAILURE	2×10^7	1	2×10^5
	OTHERS	$<2 \times 10^5$	96.5	2×10^5
	TOTAL			9×10^5
2	STEAM GENERATOR TUBE RUPTURE	1×10^7	100	1×10^7
	BYPASS VIA CONTAINMENT SUMP	-	1	-
	FAILURE TO ISOLATE	-	0.5	-
	EARLY CONTAINMENT FAILURE	-	1	-
	OTHERS	-	97.5	-
	TOTAL			1×10^7
3	STEAM GENERATOR TUBE RUPTURE WITH WATER IN SECONDARY SIDE	2×10^7	1	2×10^5
	BYPASS VIA CONTAINMENT SUMP	4×10^7	1	4×10^5
	FAILURE TO ISOLATE	2×10^7	100	2×10^7
	EARLY CONTAINMENT FAILURE	-	0.5	-
	OTHERS	-	97.5	-
	TOTAL			2×10^7

The above rough estimates are for dose within a 500 mile radius of the plant. In general, a rough estimate of dose at 50 miles may be obtained by dividing by two; however, this is a plant specific parameter which was not estimated for the Palisades plant. An uncertainty of a factor of roughly five is applicable to this parameter.

---

* Types are defined as follows:

Type 1: Loss of core heat removal with containment sprays and fan cooler available, steam line break outside of containment

Type 2: Loss of core coolant inventory subsequent to steam generator tube rupture with open steam generator relief valve

Type 3: High pressure core melt with containment bypass via broken steam line and stuck open steam generator relief valve

6.5 Procedures and Training

During the course of review of the licensee's analyses, the staff has concluded that procedural and operator training improvements are warranted to validate assumptions in the analysis and to cope with effects of some of the events as discussed below:

1) The emergency procedures dealing with secondary line breaks, EOP-6 and EOP-7, and the procedures for normal reactor trip, EOP-1, do not provide definitive guidance regarding maintenance of a heat sink (use of steam generator level or other secondary system parameter for feedback) even if the operator perceives that both steam generators may be faulted. In fact, in EOP-1 it is noted that if dryout occurs, the affected steam generator is to be considered inoperable. Additionally, the absence in the procedures of any recognition of overcooling expected to occur in these events enhances the potential for inappropriate spontaneous operator action. Procedures based on systems analysis should be developed to enable the operator to cope with these events and to achieve a controlled cooldown. The staff considers that analyses are needed to identify a control strategy that considers the decay heat generation rate, auxiliary feedwater flow requirements, instrumentation and controls available to the operator and how the response changes over the course of the cooldown.

2) Because some instrumentation inside containment may be affected by the two steam generator blowdown environment, operator training/procedures to cope with possible loss of information or misinformation is needed.

3) The ADV open automatically on high RCS temperature. The licensee noted in its evaluation that if there is a path from the RCS or the containment to the steam generator secondary system, such as would occur from steam line breaks inside containment or SGTR, a release path would exist. In the PRA the licensee assumed a high likelihood that the operators would close the valves. To validate this assumption, the procedures and training should address the need to manually close these valves to provide containment integrity under such circumstances, especially since the emphasis for the ADV in procedures may be on keeping the valves open as a heat sink.

4) A steam generator tube rupture with MSIV failure results in a loss of RCS inventory into the secondary side until system pressure is reduced. Recognition of this possibility and contingency measures should be addressed.

7.0 SUMMARY AND CONCLUSIONS

The staff has reviewed the expected response of the Palisades Plant to main steam line breaks (including valve openings) with the existing plant design. The potential improvement to safety derived from previously identified plant modifications (MSIV replacement or addition, redundant feedwater isolation) was also assessed. As discussed above, a double steam generator blowdown or a single generator blowdown with continued feedwater addition, although more severe than the licensing basis MSLB accident, is not expected to result in unacceptable consequences. Furthermore, the risk evaluation demonstrates that the potential offsite consequences are low and that the modifications would not provide a substantial improvement in plant safety.

As previously discussed, the staff's major concern for MSLB and MSIV failure was for decay heat removal capability. The staff notes that plant changes have already been implemented to improve AFW sytem reliability, such as installation of an additional motor-driven pump. The effects of addition of cold AFW to a dry steam generator have been found acceptable. Because the MSIV modification only affects AFW system reliability and equipment qualification (see below) for a limited set of initiating events, the cost-benefit evaluation indicates that the justifiable costs (averted risk) is small. Nevertheless, there are procedural improvements for operator response to steam line breaks, single or double generator blowdown, that will enhance the likelihood of successful decay heat removal, even with faulted steam lines.

Isolation of the main feedwater lines on low steam generator pressure is already provided for the main feedwater regulating valves and the bypass line valves. The bypass is open only during startup and shutdown operations. The proposed modifications to provide redundant feedwater isolation provide a benefit only for steam line breaks [inside containment] with postulated failure of the regulating valve. For the cost-benefit evaluation, the assumption was made that MSLB and feedwater valve failure results in core melt (through loss of instrumentation resulting from exceeding the qualification conditions). The results of this evaluation indicate that only a small reduction in core-melt frequency is gained by this modification.

During the PRA review, the licensee concluded that the equipment qualification envelope presently in use would be exceeded by a double steam generator blowdown or for single generator blowdowns with feedwater isolation failures. Most electrical equipment needed to respond to steam line break accidents is located outside containment; however, some instrumentation, such as steam generator level, is located inside containment. In the PRA, it is conservatively assumed that if containment conditions exceed the qualification envelope, core melt occurs. Even under this assumption, the averted risk from qualifying equipment for the environment is small. Nevertheless, the licensee has proposed to develop procedures and training to cope with loss of the instrumentation inside containment to further enhance the operators' ability to mitigate these unlikely events.

Therefore, the staff concludes that, subject to improvements in the emergency operating procedures and training (see section 6.5):

1) the existing configuration of main steam line isolation is acceptable.

2) the existing degree of isolation for the main feedwater and bypass lines is acceptable.

3) qualification of components inside containment for the temperature/pressure conditions resulting from a postulated double steam generator blowdown is not required.

Principal Contributors: Goutam Bagchi, Sanford Israel, Edward Lantz, Warren Lyon, Eileen McKenna, Jai Rajan, Paul Shemanski and Thomas Wambach.

Dated: FEB 28 1986

8.0 BIBLIOGRAPHY

1. NUREG-0820, Integrated Plant Safety Assessment Report (IPSAR) for the Palisades Plant, October 1982

2. Letter dated February 16, 1982 from R. Vincent (CPCo) to D. Crutchfield (NRC), Subject: Palisades Plant - Systematic Evaluation Program

3. IE Bulletin 80-04, "Analysis of PWR Main Steam Line Break with Continued Feedwater Addition", February 8, 1980

4. Letter dated July 26, 1983 from D. VandeWalle (CPCo) to D. Crutchfield (NRC), Subject: IE Bulletin 80-04

5. Letter dated August 15, 1983 from K. Toner (CPCo) to D. Crutchfield (NRC), Subject: Palisades-SEP Topic XV-2-Request for Time Extension

6. Letter dated September 14, 1983 from D. Eisenhut (NRC) to D. VandeWalle (CPCo), Subject: Evaluation of Main Steam Line Break

7. Letter dated May 23, 1985, from B. Johnson (CPCO) to Director (NRR), transmitting Assessment of Palisades Main Steam Line Break (MSLB) Single-Failure Backfits.

8. Letter dated November 16, 1981, from D. Crutchfield (NRC) to D. Hoffman (CPCO), Subject: Evaluation Report on SER Topics V1-2.D and V1-3.

9. CENC-1120, "Analytical Report for Consumers Power Steam Generator".

10. Letter dated April 19, 1984, from D. VandeWalle (CPCO) to D. Crutchfield (NRC), forwarding 1983/1984 Steam Generator Evaluation and Repair Program Report.

11. SECY-82-465, November 23, 1982, Pressurized Thermal Shock.

12. SECY-83-288, July 15, 1983, Proposed Pressurized Thermal Shock Rule.

13. NUREG-1044, Evaluation of the Need for a Rapid Depressurization Capability for CE Plant, December 1984.

14. NUREG-0844, NRC Integrated Program for Resolution of Unresolved Safety Issues A-3, A-4 and A-5 Regarding Steam Generator Tube Integrity, Draft Report for Comment, April 1985.

15. SECY-83-293, July 19, 1983, Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events.

16. SECY-85-163A, September 4, 1985, Station Blackout, Unresolved Safety Issue USI) A-44.

17. Letter dated October 30, 1985 from James L. Kuemin (CPCo) to Director (NRR) transmitting Component List for MSLB Evaluation.

18. Letter dated November 3, 1981 from D. M. Crutchfield (NRC) to D. P. Hoffman (CPCo) Subject: Palisades - SEP Topics - Design Basis Events.

19. Letter dated December 5, 1985 from James L. Kuemin (CPCo) to Director (NRR) transmitting MSLB Evaluation Appendix 9, Uncertainty Analysis.