Consumers Power
POWERING MICHIGAN'S PROGRESS
Palisades Nuclear Plant: 27780 Blue Star Memorial Highway, Covert, MI 49043

G B Slade
General Manager

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Nuclear Regulatory Commission
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DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT -
UNREVIEWED SAFETY QUESTION - POTENTIAL FOR LEAKAGE OF CONTAINMENT SUMP WATER TO THE SIRW TANK DURING AN MHA - REVISION 1

In a letter, dated June 14, 1991, Consumers Power Company (CPCo) notified the NRC that we had determined that a potential leak path existed whereby previously unaccounted for radioactive post accident primary coolant system (PCS) water could leak to the Safety Injection and Refueling Water Tank (SIRWT). The tank is vented directly to the atmosphere and the SIRWT as a dose contributor had not been identified as a source term when calculating the control room or off-site doses in accordance with General Design Criteria (GDC) 19 and 10 CFR 100. CPCo notified the NRC that this situation had been reviewed and considered as an unreviewed safety question. In accordance with 10 CFR 50.59 and pursuant to 10 CFR 50.90, CPCo requested that the Palisades Facility Operating License be amended by granting an exemption for the FSAR requirement to perform the Maximum Hypothetical Accident (MHA) analysis in accordance with the Standard Review Plan, Section 15.6.5, Appendix B, Subsection II(1), which specifies that leakage as a result of passive component failure is included in determination of the radiological consequences of a design basis loss-of-coolant accident. We further requested this exemption be approved until the beginning of Cycle 10, which we believed would allow us time to complete a revised MHA analysis and necessary modifications to allow the plant to meet the passive component failure criteria of the Standard Review Plan (SRP). The June 14, 1991 letter described general actions being taken to resolve the concern and an attachment to the letter contained a description of the issue, a justification for continued operation and a no significant hazards determination.

Following further reviews CPCo determined that a commitment, in the June 14, 1991 letter, was inappropriate. A supplemental letter providing a revised commitment was submitted on July 17, 1991.

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Consumers Power Company's continuing evaluation of the MHA analysis has revealed that certain other analysis inputs need to be reconciled to the actual plant configuration or updated in order to meet the present SRP methodology. Based on the scope of analysis inconsistencies, we anticipate that plant modifications may be required to update the plant design to be consistent with the present SRP methods and requirements. In order to determine to the fullest extent what modifications may be necessary, a thorough re-documentation and calculation of our MHA analysis is required. The MHA analysis will necessarily have to consider future modifications are implemented. Even though this work is ongoing, it is anticipated that the final MHA results will not be available until the end of the first quarter of 1992. This will not allow enough time for NRC review and time to implement any resultant modifications prior to start-up from the Cycle 10 refueling outage.

Therefore, CPCo requests that our previous amendment request submitted in our June 14, 1991 letter be revised as described below. In accordance with 10 CFR 50.59 and 50.90, CPCo requests that the Palisades Facility Operating License be amended by granting an exemption from the FSAR requirement to perform the MHA analysis in accordance with the Standard Review Plan section 15.6.5. It is requested that the exemption remain in place until no later than start-up for the beginning of Cycle 12. This time interval will be needed to plan and implement plant modifications to support the revised MHA analysis. We anticipate that as a result of our reviews we will be changing our method of neutralizing the containment sump water following a loss of coolant accident. Modifications to the control room ventilation system may also be required.

CPCo will submit the revised MHA and control room habitability analysis to the NRC for review and approval. Following NRC approval, CPCo will proceed with installation of the defined modifications. It is anticipated that the new analyses will be submitted to the NRC by April 30, 1992. If the NRC can issue an SER by November 1, 1992, CPCo will complete all identified modifications by the end of the 1994 refueling outage (beginning of Cycle 12).

Attached is a revision to the information we provided with the June 14, 1991 submittal which contained a description of the issue, operability determination, and a no significant hazards determination. Based on previous analysis, previous NRC safety evaluation reports, relative risk comparisons of an MHA occurrence versus the Palisades plant's other accident scenarios and the information provided with this correspondence, we believe the plant's continued operation will present no significant increased risk to the health and safety of the public.

Preliminary evaluations show that off-site doses and control room whole body doses would not exceed prescribed limits. Control room thyroid doses are the main concern. However, potassium iodide thyroid blocking tablets are available and provisions exist in the emergency implementation procedures for its issuance.

Changes to the June 14, 1991 submittal are indicated by a vertical line in the right hand margin, opposite the change.

Gerald B Slade
General Manager

CC Administrator, Region III, USNRC
NRC Resident Inspector - Palisades

Attachment

The Palisades Plant Review Committee has reviewed this Amendment Request and has determined that this change does involve an unreviewed safety question. The change involves no significant hazards consideration. This change has been reviewed by the Nuclear Performance Assessment Department. A copy of this Amendment Request has been sent to the State of Michigan official designated to receive such Amendments to the Operating License.

CONSUMERS POWER COMPANY

To the best of my knowledge, information and belief, the contents of this submittal are truthful and complete.

By _____
David P Hoffman, Vice President
Nuclear Operations

Sworn and subscribed to before me this 8th day of January 1992.

_____ [SEAL]
LeAnn Morse Notary Public
Van Buren County, Michigan
My commission expires June 6, 1994

ATTACHMENT

Consumers Power Company
Palisades Plant
Docket 50-255

UNREVIEWED SAFETY QUESTION-POTENTIAL FOR LEAKAGE OF CONTAINMENT SUMP WATER TO THE SIRW TANK DURING AN MHA
REVISION 1

14 Pages

1.0 INTRODUCTION

Palisades has been evaluating a possible unanalyzed release path through an atmospheric vent located on top of the Safety Injection Refueling Water (SIRW) Tank. Release scenarios assume that if a LOCA were to occur the MHA source terms would exist. Therefore, following recirculation actuation and alignment of engineered safety feature (ESF) pumps to the containment sump, pressure in the discharge header, combined with assumed SIRW Tank isolation valve leakage, would cause back flow into the SIRW tank which could be released directly to the atmosphere through the unmonitored 6-inch tank vent. Earlier analysis of a similar line on the suction side of the ESF pumps had concluded that this was not a concern due to the high elevation of the tank. Continued work on the issue, however, identified two additional lines downstream of the safety injection pumps which could also return water to the SIRW tank. If valve leakage exists, the uncontrolled and unmonitored release through the SIRW vent becomes a concern.

As reevaluation of the MHA work has progressed, additional questions have been raised about MHA analysis assumptions for iodine retention in containment sump recirculated water and about control room habitability analysis (CRHAB).

2.0 SYSTEM DESCRIPTION

SIRW Tank Leakage

The high and low pressure safety injection pumps and the containment spray pumps take suction from the SIRW tank through two separate 18 inch suction headers. Each header is isolated during a recirculation actuation signal (RAS) from the SIRW tank using one air operated isolation valve and one check valve in series. The same valve configuration also isolates the two containment sump headers. To provide proper suction pressure to the pumps, the SIRW tank is located on top of the auxiliary building roof approximately 75 ft above the pump suction inlet.

Discharge of the containment spray pumps flows through the shutdown cooling heat exchangers and into the containment building atmosphere. Discharge of the high and low pressure safety injection pumps is directly into the primary coolant system. Two lines are included downstream of these pumps which return to the SIRW tank. The first line is the minimum flow path from each ESF pump which combines into a single header and is returned to the SIRW tank. When RAS occurs, this line is isolated by two independent isolation control valves (CVs) in series. The second line originates downstream of the shutdown cooling heat exchangers and is used for mixing of the SIRW tank and for system testing. This test line is isolated from the SIRW tank during operation by a locked closed manual isolation valve (MV).

Control Room Ventilation

As described in FSAR Sections 6.10, 5.1, 1.8 and 9.8, the control room ventilation system functions to prevent air in-leakage during normal and post-accident conditions by maintaining a minimum positive air pressure of .125 inches of water greater than all adjacent areas. During post accident operation, the system removes (by charcoal filtration) airborne radioactive iodine from the control room and the outside air makeup to ensure habitability of the control room. In 1983 and 1984 responses to commitments made for compliance to NUREG-0717 III.D.3.4, "Control Room Habitability Requirements," the system was modified to extend the emergency control room air intake from the then existing configuration, increase the intake air duct to allow 100% makeup air, install redundant charcoal filters, extend the control room habitability zone and replace air intake and discharge dampers.

The system operates in either the normal, emergency or purge mode. During the normal mode, one of the system's two trains is in operation.

The emergency mode of operation is activated either by a containment high pressure signal, containment high radiation signal or manually from the control room. During the emergency mode, the air handling units and charcoal filter units in both trains operate. The control room operator has the option to turn one of the trains off.

During the emergency mode of operation the control room exhaust air is conditioned and returned to the control room. However about one third of this air is drawn from the emergency outside duct.

Iodine Removal System

As described in FSAR Sections 5.1 and 6.4 the iodine removal system acts in conjunction with the containment spray system to reduce the post-accident level of fission products in the containment atmosphere. The iodine removal system consists of a hydrazine tank and a sodium hydroxide tank located adjacent to the SIRW tank on the roof of the auxiliary building. Hydrazine and sodium hydroxide solutions are fed through two sets of parallel headers to the suction headers of the low-pressure and high-pressure safety injection pumps and containment spray pumps.

The original plant system design provided for the automatic addition of sodium hydroxide (NaOH) to the water from the SIRW tank after a LOCA, to provide for both iodine retention and neutral pH control. Pursuant to NRC direction, Amendment 31 to the operating license and changes made in support of Amendment 40 to the operating license, the iodine removal system was modified to provide for automatic addition of hydrazine rather than Sodium Hydroxide (NaOH) for purposes of iodine retention. The sodium hydroxide was retained, however, for long-term control of pH.

The hydrazine tank contains 270 ± 17 gallons of 15.5 ± 0.5% (by weight) hydrazine solution with a nitrogen cover gas pressure of 11.2 ± 2 psig.

The sodium hydroxide tank, is available for continued pH control during the recirculation phase of the DBA. It provides a storage volume of 4,000 ± 300 gallons of 30.0 ± 0.5% (by weight) sodium hydroxide solution with a nitrogen cover gas.

Upon receipt of a containment high-pressure signal, the power-operated valves in the chemical injection lines from the hydrazine tank will be opened after a one-minute time delay. The one-minute time delay provides the operator time to prevent inadvertent opening of the chemical injection valves which would result in the spraying of the hydrazine solution on equipment or workers in the containment building when it is not required; such as, from a spurious operation or secondary steam breaks where no clad failure has occurred.

After injection and mixing of the hydrazine is complete, the composition of the mixed solution will be maintained at a pH of approximately 7.0 by manually adding NaOH from the sodium hydroxide tank into the engineered safeguards pump suction. Sampling connections are available to periodically examine containment sump water pH during recirculation and make adjustments as required.

3.0 BACKGROUND

SIRW Tank Leakage

In 1987, a special test procedure was developed with the intent of measuring the leakage through the SIRW tank isolation valves. Although analysis showed the leakage through the SIRW tank outlet isolation valves was not likely to cause back flow into the SIRW tank due to the elevation difference between the SIRW tank and the containment sump (following RAS), the procedure was developed at the request of the independent safety review group. The test was first performed in September of 1990 at which time the valves failed their leak rate acceptance criteria. Root cause evaluation was initiated and again concluded that back leakage through the SIRW outlet isolation valves was not a concern due to the high elevation of the SIRW tank. However, this root cause evaluation raised the concern of other possible leak paths when considering the recirculation and test headers. To address the concern, further analysis was performed and concluded that a concern indeed existed.

Control Room Ventilation

On November 1, 1977 the NRC issued an SER for Technical Specifications Amendment 31, which allowed Palisades to increase power from 2200 MWt to 2530 MWt. The NRC stated that a control room habitability analysis had not been addressed in the FSAR or by the NRC, and noted they would resolve the matter (compliance with GDC-19) prior to Cycle 3. In an NRC letter dated May 4, 1978, the NRC moved the requirement to resolve the lack of analysis to verify control room habitability to the Systematic Evaluation Program (SEP). In October 1982, NUREG-0820 was issued by the NRC giving the results of the Palisades SEP. NUREG 0820 listed control

room habitability as deleted from SEP review because NUREG-0737 Topic III.D.3.4 was identical. On April 29, 1983 the NRC transmitted the SER for NUREG-0737 Topic III.D.3.4 accepting CPCo's proposed control room ventilation design as meeting the review requirements after modifications, with four areas recognized as departures from the Standard Review Plan. The four areas were 1) the remote air intake is not seismic and missile protected; 2) radiation and smoke detectors not at the intake; 3) air conditioning compressors are not provided with automatic diesel power in the event of loss of off-site-power; 4) some instrumentation useful in confirming isolation of the normal intake by redundant dampers not provided in the control room.

A new control room heating, ventilating and air conditioning system was installed and operational in June of 1984.

Iodine Removal System

Under original plant design reviews, CPCo proposed to add a sodium hydroxide iodine removal system. The NRC approved plant operation at the 2200 MWt level until this issue could be resolved. Various studies were completed by the NRC to support the use of sodium hydroxide. Amendment 22 to the Palisades Construction Permit and Operating License, dated May 28, 1971, validated the use of sodium hydroxide to lower dose consequences. Supplement 3 (June 12, 1971) to the AEC Original SER determined that sodium hydroxide mixed with SIRWT Tank injection would make the containment sump liquid a pH of 7. On January 22, 1974, CPCo's Full Term Operating License Application, Amendment 28, was sent to the NRC and concluded that a 6.4 minute delay in injecting sodium hydroxide would be acceptable to prevent inadvertent spraying of containment with a NaOH solution.

CPCo's letter of April 20, 1976 requested a delay in placing the iodine removal system into operation in order to complete further analysis. A CPCo October 7, 1977 letter noted that all analysis was completed and committed to placing the iodine removal system into service prior to any power operation above 2200 MWT or the next refueling outage.

CPCo's October 25, 1977 letter agreed with the NRC staff recommendation to lower Engineered Safety Features (ESF) leakage and change from sodium hydroxide to hydrazine during SIRW Tank injection, to provide a 75 ppm concentration in the containment spray. At this point the design of the system was finalized. Various analysis revisions by CPCo and the NRC have since been made over the years to upgrade and revise the analysis conclusions.

4.0 CURRENT STATUS

SIRW Tank Leakage

Preliminary calculations show that a very low leak rate through the SIRW tank isolation valves will approach or exceed control room limits set in GDC-19 and site boundary limits set in 10 CFR 100. The leakage past the

isolation valves cannot be measured until plant modifications are made.

Analysis and information concerning this issue has been developed into a justification for continued operation and reviewed by the Plant Review Committee (PRC) with the conclusion that an unreviewed safety question currently exists. Because the exact amount of leakage from the valves in question is not known, it was decided that when excessive valve leakage is considered, the consequences of an accident or a malfunction of equipment important to safety could be increased beyond what had been previously considered in the FSAR. The control valves are already the subject of a docketed relief request, pending modifications in 1992, to provide leak testing capability. Modifications were being considered to provide for filtering of the SIRW tank vent.

Consumers Power Company's July 17, 1991 letter stated we had determined that, instead of installing a charcoal filter in the SIRW Tank vent line to minimize the possible radioactivity leak following an MHA, a new valve line up which directed leakage to the spent fuel pool, has been implemented. Modifications are still planned for this refueling outage to allow leak testing of the valves.

We are continuing with the following actions for resolution of this concern.

1) Complete modifications to allow seat leakage testing of the control valves by the next refueling outage.

2) Evaluate a modification to manual isolation valve MV-3225 which will allow testing or ensure a leak tight barrier.

3) Continue to evaluate other appropriate actions which would temporarily or permanently eliminate the consequences of leakage or eliminate the leak path.

Establishment of definitive acceptance criteria for valve leak rates, however, may not be possible without resolution of all iodine removal, control room habitability and MHA issues discussed below.

Control Room Ventilation

The source of radioactivity which is input to the CRHAB analysis is that activity calculated to be released in the MHA analysis.

Because of the concerns identified regarding a potential dose contribution from the SIRW Tank and the potential for changes in the MHA analysis results, the Control Room Habitability Analysis (CRHAB) was reviewed to assure compliance with applicable dose criteria. While some problems have been discovered with the analysis methodology, the biggest effect on the analysis stems from the input source activity we have to assume is released from the SIRW tank and containment.

We are continuing a verification of the CRHAB parameter inputs against the physical design requirements to assure the inputs are accurate. Final confirmation, however, must await completion of the revised MHA analysis.

In the interim, because of the uncertainty in analysis inputs, we will provide guidance in the emergency response procedure regarding the issuance of Potassium Iodide (KI) tablets.

Iodine Removal System

The most significant concerns about the MHA analysis relate to assumptions about iodine retention in the sump water. Iodine, which is airborne in containment following the MHA, is assumed to be released directly to the atmosphere via containment leakage. Iodine retained in containment sump water can be released through direct leakage to the atmosphere and indirect leakage via ESF piping and the SIRWT. If a portion of the iodine is not retained in water (or plated out on containment structures), it must be assumed available to leak out of containment with other airborne activity. Changes in release path assumptions can cause different predicted doses to the general public or control room operators. For Palisades, if a greater portion of the iodine is assumed in the containment atmosphere, predicted doses to the general public and control room operators would increase. However, with more of the iodine in the containment atmosphere, off-site dose is still not expected to exceed 10 CFR 100. This is because doses predicted by current analyses are substantially less than the limits allowed by 10 CFR 100. Calculated control room operator doses, on the other hand, could exceed the GDC 19 limit of 5 rem whole body equivalent because the currently calculated dose is near the limit.

Iodine retention in the sump water has been brought into question for two reasons. First, the NRC guidelines regarding the value of hydrazine as an iodine scavenging spray additive have changed since original licensing of the systems in the late 1970's. The Standard Review Plan, 6.5.2 Rev 2, now credits fresh water, containing no dissolved iodine, as being just as effective for iodine scavenging as spray solutions containing trace levels of a chemical additive. Second, current NRC guidance specifies that a sump pH of 7 or above should be established by the time recirculation begins. A pH in this range ensures that the iodine scavenged from the atmosphere is retained in the sump and does not evolve back out of solution. The Palisades system configuration and procedures do not guarantee the desired pH can be achieved until several hours after a LOCA. The evolution of iodine out of low pH sump water has never been explicitly addressed in either NRC or CPCo analyses of the MHA for Palisades.

5.0 Evaluation and Management of Integrated Issues

Palisades Corrective Action document, E-PAL-90-035, documented that, during recirculation following a LOCA, highly contaminated containment sump water could enter the SIRW tank and contribute to airborne release of radioactive materials. If this leakage through CV-3056, CV-3027, and/or MV-3225 were accounted for, doses calculated by the MHA and CRHAB analyses could be increased. A June 1991 evaluation of this issue led to the estimate that leak rates greater than 2 gpm could result in exceeding MHA dose limits in 10 CFR 100, and leakage greater than .5 GPM could cause CRHAB limits in GDC 19 to be exceeded. This was reported to the NRC as an Unreviewed Safety Question in a letter dated June 14, 1991. An exemption from Licensing Bases was requested until leak testing of the valves could be performed and MHA and CRHAB analyses revised to define leakage acceptance criteria.

As review work has progressed, additional questions have been raised about MHA analysis assumptions for iodine retention in recirculated sump water and about CRHAB analysis methodology. The Plant Review Committee (PRC) met on November 1, 1991, and discussed the implications of this information. Using the guidance of the August 9, 1989 NRC internal memo from Jim Partlow, the issue of operability of the iodine removal and control room ventilation systems was discussed. The Technical Specifications Section 1.4 definition of operable states that "A system or component is operable if it is capable of fulfilling its design function." Since the iodine removal system can function as designed and is licensed with manual addition of sodium hydroxide after the start of recirculation, the system was determined operable. PRC also discussed operability of the control room ventilation. One of its design functions is to maintain the control room habitable so that the calculated dose to operators over the 30 days following a LOCA would be less than 5 rem whole body or its equivalent to any part of the body. In broad terms, the inability of the system to achieve dose limits could be construed as being inconsistent with the Technical Specifications definition of operability. PRC also noted, however, that the analysis which calculates operator doses has a number of conservative assumptions which are not considered realistic. For example, the analysis assumes that the operator occupies the control room continuously for the first 24 hours after the LOCA without shift rotation. It assumes containment leakage is at a design 0.1% per day by weight regardless of the known actual leakage rate. It assumes a very conservative source term which is based on a conservative power history in the MHA analysis. In addition the analysis is not a calculation of a short term response to a crisis situation which relies on equipment, but is instead a prediction of dose accumulated over 30 days. PRC felt confident that over a 30 day period many actions would be available to limit operator doses. In fact control of operator doses would not rely simply on system design, but would rely on management of stay times, use of protective equipment if unexpected conditions were experienced, and potentially use of KI tablets.

With this understanding of the conservative nature of MHA and CRHAB analyses, PRC concluded that the control room ventilation system was

still capable of fulfilling its design function of limiting predicted operator doses to 5 rem over the 30 days after a LOCA. PRC also directed that an in depth review of the situation be completed to determine whether additional short term measures might be in order. Possible measures to consider might include; 1) specifying and/or taking analytical credit for limited operator stay times, 2) administratively limiting (and/or taking analytical credit for) actual containment leakage known to be well below the NRC operating limit which is itself below the total leak rate assumed in the analysis, and 3) revising operating procedures to add sodium hydroxide earlier after the accident without waiting for a chemistry sample.

PRC also concurred with the recommendation to submit a voluntary report to the NRC under 10 CFR 50.72.

SUBSEQUENT INVESTIGATION

Since the November 1, 1991 PRC meeting, further investigation has been conducted to better define the licensing bases for CRHAB and MHA, so that the full implications of the questions could be assessed, and to explore the practicality of possible short term actions which could bring the plant into compliance if it were determined to be necessary.

The investigations revealed that the licensing bases regarding MHA and CRHAB were not clear. When the licensed power level was increased to 2530 MWt, the MHA analysis of record became the one performed by the NRC and documented in the SER for Amendment 31 to Technical Specifications, dated November 1, 1977. This analysis assumed credit for hydrazine as an enhanced airborne iodine removal mechanism instead of NaOH. No treatment is provided for possible evolution of iodine out of sump water before initiation of NaOH addition. On March 9, 1978 CPCo submitted its own MHA analysis in support of a proposed Technical Specifications Amendment. In this submittal, it was stated that NaOH addition could be initiated in one hour after the LOCA, but no credit was taken for an increased iodine removal coefficient until 2-1/2 hours after the event. The NRC SER, dated April 12, 1978, for Amendment 40 to Technical Specifications only stated that "... the [March 9, 1978] proposal provides adequate assurance that the iodine removal rates assigned the Palisades iodine removal system in the SE for Amendment 31 is justified." In short, both NRC and CPCo gave credit per the guidance at the time for improved iodine removal due to hydrazine addition to the spray water, and neither addressed the evolution of iodine back out of the sump water when pH was less than 7.

When the updated FSAR was developed and submitted to the NRC in 1984, it included the NRC MHA analysis results from Amendment 31 as the analysis of record. Over subsequent years, updates to the MHA analysis and development of new analyses for CRHAB based on MHA analysis results were performed and documented. All carried forward the original NRC credit for hydrazine removal of iodine, and all continued the assumption that iodine would remain in solution until the time NaOH was added for pH adjustment. Other methodology and assumptions, however, were not consistently carried forward to all later analyses. One change of

specific note is the adoption of International Commission on Radiological Protection (ICRP) Report 30 methodology to calculate control room whole body equivalent doses. ICRP 30 was issued in 1978 but it has not been formally adopted by the NRC for accident analyses. It was, however, recently (June, 1991) adopted as the basis for occupational exposures under the new 10 CFR 20. While CPCo notified the NRC in LER 88-013, Rev 1 dated November 17, 1988 that ICRP 30 was being used to calculate whole body doses, formal approval of ICRP 30 has not been requested nor received.

The investigation also confirmed that some actions were available to operating personnel to reduce off-site and control room operator doses following the MHA. For example, manual operator action could be taken more promptly than now specified in Emergency Operating Procedures to manually add NaOH to the sump. Imposing additional actions such as this on operators, however, conflicts with our general desire to avoid short term manual actions in accident situations unless absolutely necessary. From a safety standpoint it does not appear necessary to require additional immediate actions to limit control room or off-site doses.

A review of current criteria for performance of MHA analyses was also of little value in trying to clarify current Palisades licensing bases. Current methodology and standard assumptions embodied in the current Standard Review Plan are different from those of 1977 and 1978. As these criteria have changed, however, the MHA and control room habitability analyses have not been updated. This suggests that adequate protection is believed to exist even though the state of knowledge has advanced and the specific margin has not been identified by the most current techniques.

6.0 PROPOSED PLAN FOR RESOLUTION OF ISSUES

CPCo proposes to provide an integrated resolution for all issues by developing a new MHA and control room habitability analysis using current NRC criteria, but not necessarily current plant design. This plan envisions that the new analysis would incorporate all design and license changes anticipated to be necessary to achieve compliance with current NRC criteria. Changes that might be incorporated, for example, include elimination of hydrazine, installation of passive pH control devices in the containment sump, definition and inclusion of realistic valve leak rates into the SIRWT, possible control room ventilation system modifications, etc. Implicit in this approach is a commitment to modify the plant if the analysis results show the need and cost effectiveness of those modifications. The previous SRP departures approved by the NRC's resolution of Nureg-0737 Topic III.D.3.4 for the Palisades plant, will remain as is.

This approach would not give a definitive answer about compliance with the current poorly defined licensing bases. It would, however, provide a substantial upgrade in the physical plant's ability to mitigate a major LOCA, it would clearly define applicable design and licensing bases for the future, and it would result in MHA and control room habitability analyses which clearly document compliance with those design and licensing bases.

This approach is considered to be a far better use of resources than attempting to reconstruct old analyses using outmoded methodology to document compliance with criteria no longer acceptable for newer plants.

It is proposed that CPCo would submit the MHA and CRHAB analyses of the "reconfigured" (if necessary) plant to the NRC for review, approval and issuance of an SER. Following NRC approval, CPCo would proceed with installation of the defined modifications. It is anticipated that the new analyses could be submitted to the NRC by April 30, 1992. If the NRC can issue its SER by approximately November 1, 1992, CPCo would expect to complete all modifications by the end of the 1994 refueling outage.

7.0 ACCEPTABILITY OF CONTINUED PLANT OPERATION

In spite of some uncertainty with current plant analyses, plant operation prior to completion of analyses and associated modifications is judged to be safe and acceptable, and would create no undue risk to the public or control room operators. It is understood that all modifications determined to be necessary may not be installed until the 1994 refueling outage. The following discussion provides the bases for this judgement:

A. The MHA analysis is not a realistic prediction of the impact of a major LOCA. It is, instead, an extremely conservative bounding calculation. As stated in Palisades FSAR Section 14.22.1,

> "The Maximum Hypothetical Accident (MHA) is postulated to release substantially more fission products and result in more severe consequences than any incident considered credible. The evaluation is only meant to determine a reasonable upper bound of the consequences of an incident involving the release of radioactive material from the plant site. The radiological consequences of the MHA are determined in a manner independent of any specific plant transient sequence that might be postulated. To this end, the evaluation is performed in accordance with the guidelines and recommendations put forth by the NRC staff. These guidelines are nonmechanistic in nature and are meant to maximize the consequences of the MHA."

In addition to the conservative MHA results, control room habitability analysis methodology adds additional conservative assumptions. In this context, even if a control room habitability analysis showed operator doses above five rem, actual doses could be expected to be considerably less.

B. FSAR Section 14.22 discusses the current MHA analysis for Palisades. The site boundary and low population zone thyroid doses calculated in this analysis are 69.72 rem and 39.63 rem respectively (as compared to the 10 CFR 100 limit of 300 rem). Because of this large margin, off-site exposures from the MHA are judged not to be of concern, even with the questions about methodology and assumptions discussed above.

C. Two sources of conservatism in the MHA analysis are worthy of particular note because they directly impact calculated control room doses. First, the MHA analysis assumes instantaneous release of fission products from the fuel at the time of the LOCA. 100% of the total core inventory of iodine and noble gasses is assumed to be released to containment. Second, the MHA analysis assumes that containment leakage is at a full .1 weight percent (w/o) per day for 24 hours and .05 w/o per day thereafter for 30 days regardless of containment pressure. In reality the leakage would be significantly less, especially in later stages of the accident due to the reduced containment pressure. Even the value of .1 w/o per day is conservative. 10 CFR 50 Appendix J specifies that the ILRT leak rate acceptance criterion as .075% per day by weight. Palisades Technical Specification 4.5.2 b(1) specifies the containment leakage limit for local leak rate tests as .6La or .06% per day by weight. Current actual leak rate as measured by the last ILRT and monitored by the LLRT program is approximately .04% per day.

D. The control room habitability analysis uses the same MHA releases that contribute to off-site dose to calculate a theoretical dose to control room operators. One extremely conservative assumption in this calculation has the operator stationed in the control room and exposed to this radioactivity for 100% of the first 24 hours, 14.4 hours per day for the next three days, and 10.4 hours per day for the next 26 days straight. It also assumes that no protective measures such as use of respirators and potassium iodide are taken. Actual exposure of control room operators to whatever control room activity might be present can be expected to be considerably less, if only because there are five operating crews which would be on rotating shifts during this period.

The following comments were provided in the June 14, 1991 letter and pertain specifically to the acceptability of continued Plant Operation without the ability to measure the leak rate through SIRW Tank isolation valves. They are restated here for completeness.

1) The fuel damage and release of fission products to the containment that are assumed in the MHA analysis are much larger than one would expect for any of the Palisades Design Basis Accidents. The Palisades FSAR (Section 14.22.1.1) states, "The Maximum Hypothetical Accident (MHA) is postulated to release substantially more fission products and result in more severe consequences than any incident considered credible." The large break LOCA would be expected to give the largest release of fission products to containment for the Design

Basis Accidents, but the LOCA analysis of record is performed to meet the acceptance criteria of 10 CFR 50.46 and does not determine dose consequences. The FSAR Design Basis Accident with the greatest radiological consequences, other than the MHA, is the Control Rod Ejection. The Control Rod Ejection analysis for Cycle 9 predicts approximately 15% fuel damage which is assumed to release the gap fission products. In considering the Control Rod Ejection fuel failures, the maximum leakage rate to the SIRW tank, that would still maintain control room doses below GDC-19 limits, is approximately 9 gpm, and approximately 18 gpm to maintain site boundary doses below 10 CFR 100 limits.

2) The present evaluation for determining the allowable leak rate for these valves has several simplifying assumptions that will be eliminated in a more detailed analysis. One assumption is that the leakage is transported directly from the sump to the atmosphere. This assumption neglects any fission product reduction due to decay or removal due to having to travel through a lengthy section of pipe into the SIRW tank, mixing with the volume of air in the tank, and then venting out of the tank. For example, it would take over an hour for the fission products leaking past the valves at 1 gpm to reach the SIRW tank.

3) The probability of an accident causing fuel damage and a large fission product release to containment is very small. Preliminary PRA sequence quantification indicates that a large LOCA core melt frequency would be on the order of 10^{-6} occurrences per year.

4) A large break LOCA that would cause fuel damage would also mean that the PCS boundary is not intact and the HPSI and LPSI pumps would have discharge pressures that are at a minimum (HPSI at approximately 400 psig). This would reduce the driving head for the leak through control valves to a minimum. Conversely, accidents where the PCS pressure remains high would be expected to produce less fuel damage and minimum fission product release to the containment sump.

5) The control valves are in series gate valves such that both would have to leak to produce a release. These valves are normally open and are seldom operated. These facts combined with the use of non-corrosive materials and SIRW tank chemistry control, indicate that no significant mechanism exists for valve degradation. The leak rate through these valves can be reasonably expected to be quite small. In addition, during the past refueling outage, the control valves were disassembled and refurbished and reassembled. Maintenance included seat blueing tests and stroke testing. Thus, the inability to test valve leakage is not expected to have serious safety implications.

6) The manual isolation valve (MV) 3225 was refurbished including a seat blueing test in 1988. This valve is opened only during testing of the containment spray pumps during a refueling outage surveillance. Although no specific valve leakage is known, the total PCS leak rate

when on shutdown cooling, during the 1990-1991 refueling outage, when a higher pressure exists at the MV than would occur during the containment sump recirculation mode, was observed to be less than 1 gpm. Little if any of this total PCS leak rate is thought to be due to MV 3225.

7) The dose in the control room can be reduced by a factor of approximately ten by issuing potassium iodide tablets to block the intake of radioactive Iodine to the thyroid. This would increase the allowable leak rate for the control room dose analysis, if we concluded that early post-LOCA action to distribute potassium iodine tablets should be taken.

8.0 DETERMINATION OF NO SIGNIFICANT HAZARDS

Continued operation of the plant does not involve a significant increase in the probability of an accident previously evaluated because the presence of leakage from the safeguards pumps in the small amounts of concern cannot cause or influence the probability of an accident.

The calculated consequences of an accident are potentially increased by leakage through the valves to the SIRW tank and by changes in analysis assumptions. Given the foregoing discussion, the actual consequences of any Design Basis Accident, except the MHA, are expected to remain within acceptance limits. The calculated consequences of the MHA are not expected to exceed dose limits for the general public. The calculated consequences could exceed the dose limits for plant control room operators but not significantly. The conservatism that exist in the present MHA and control room habitability analyses, combined with the simplifying conservative assumptions made in determining the effect of the SIRWT valve leakage, are believed to result in an overestimation of the actual MHA and control room doses that a detailed calculation will determine. Also, there is no indication or expectation that gross leakage does exist through these SIRWT valves.

The inability to measure the leakage through these SIRWT valves or the possibility that small leakage might exist does not create the possibility of a new or different kind of accident from any accident previously evaluated. Changes in analysis assumptions also do not create new accident types.

The inability to measure the leakage through these valves or the possibility that small leakage might exist could possibly reduce the margin of safety. Different analysis assumptions and/or methodology can also change perceived margins of safety. The calculated doses could be increased by the additional radioactivity assumed to be released from the SIRW tank and by analysis assumption changes but the increase would not be a significant safety concern because of the reasons previously stated. The actual consequences from all of the Design Basis Accidents are expected to be below limits. The allowed valve leakage can be increased by removing conservatism in the analysis and there is no reason to

believe that gross leakage exists. Although uncertainty is introduced by an imperfect knowledge of the licensing bases for MHA and control room habitability analyses, and by the licensed hydrazine system we do not believe that 10 CFR 100 limits will be exceeded. Therefore, continued operation of the plant with the exact leakage rate of these valves unknown and existing analysis uncertainties does not represent a significant reduction in the margin of safety.