NUREG–1424

Safety Evaluation Report

related to the full-term operating license for Palisades Nuclear Plant

Docket No. 50–255

Consumers Power Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

November 1990

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ABSTRACT

The Safety Evaluation Report for the full-term operating license application filed by the Consumers Power Company for the Palisades Nuclear Plant has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Van Buren County, South Haven, Michigan. The staff has evaluated the issues related to the conversion of the provisional operating license to a full-term operating license and concludes that the facility can continue to be operated without endangering the health and safety of the public.

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

ABBREVIATIONS

ANSI	American National Standards Institute
ARI	alternate rod injection
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
B&W	Babcock & Wilcox
BWR	boiling-water reactor
CE	Combustion Engineering, Inc.
CFR	Code of Federal Regulations
CMAA	Crane Manufacturers Association of America
CPCo	Consumers Power Company
DOR	Division of Operating Reactors
EDG	emergency diesel generator
EQ	environmental qualification
FES	final environmental statement
FTOL	full-term operating license
GDC	general design criterion
GIP	Generic Implementation Procedure
GL	generic letter
GSI	generic safety issue
IEEE	Institute of Electrical and Electronics Engineers
IPE	individual plant examination
IPSAR	Integrated Plant Safety Assessment Report
LBB	leak before break
LLNL	Lawrence Livermore National Laboratory
LOCA	loss-of-coolant accident
LTOP	low-temperature overpressure protection
MCC	motor control center
MPA	multiplant action
NEPA	National Environmental Policy Act of 1969
NRC	Nuclear Regulatory Commission
NUMARC	Nuclear Management and Resources Council
POL	provisional operating license
PORV	power-operated relief valve
PTS	pressurized thermal shock
PWR	pressurized-water reactor

RG	regulatory guide
RPS	reactor protection system
RPV	reactor pressure vessel
SE	safety evaluation
SEP	Systematic Evaluation Program
SER	safety evaluation report
SLCS	standby liquid control system
SQUG	Seismic Qualification Utility Group
SRP	Standard Review Plan
SSE	safe-shutdown earthquake
STS	Standard Technical Specifications
TMI-2	Three Mile Island Unit 2
TS	Technical Specifications
USE	upper-shelf energy
USI	unresolved safety issue

1 INTRODUCTION AND DISCUSSION

1.1 Introduction

This Safety Evaluation Report (SER) relates to the proposed issuance of a full-term operating license (FTOL) for the Palisades Nuclear Plant (Palisades or the plant) in response to an application filed by the Consumers Power Company (CPCo or the licensee). This report was prepared by the staff of the Nuclear Regulatory Commission (NRC or the Commission) to summarize the results of staff reviews of the proposed conversion of the provisional operating license (POL) to an FTOL.

Between 1959 and 1971, the Atomic Energy Commission (a predecessor agency to the Nuclear Regulatory Commission (NRC)) issued POLs for 15 power reactors. These POLs were for periods up to 18 months to allow an interim time of routine operation during which both the licensee and the NRC staff (the staff) could assess plant operating parameters against predicted or design values and resolve any generic concerns identified during the licensing process.

The POL for Palisades was issued on March 24, 1971, and was due to expire on March 1, 1974. However, on January 22, 1974, CPCo applied for conversion of the POL to an FTOL, thus, pursuant to the provisions of Section 2.109 of Title 10 of the Code of Federal Regulations (10 CFR 2.109), continued operation of the facility beyond the stated license expiration date is authorized pending disposition of the timely application by the NRC. The plant achieved initial criticality on May 24, 1971, and entered commercial operation on December 31, 1971.

Because of the large number of unresolved generic issues relevant to the operation of those plants operating under POLs, the staff stopped reviewing POL conversions in 1975, and, instead, set out to establish the appropriate review scope to support license conversion. Since much of the review necessary for POL conversion was similar to the scope proposed for the Systematic Evaluation Program (SEP), the staff recommended to the Commission in 1977 that POL facilities be included in Phase II of the SEP. That recommendation was adopted.

The SEP was conceived in recognition that adequate documentation is necessary to substantiate the staff's position that those plants authorized to operate early in the evolution of the NRC's licensing requirements are acceptably safe. The objectives established for the SEP were identified as

(1) The program must assess the safety adequacy of currently licensed plants.

(2) The program must document how each operating plant compares with current criteria on significant safety issues, and should provide a rationale for significant departures from these criteria.

(3) The program should provide the capability to make integrated and balanced decisions with respect to any required backfits.

(4) The program should be structured for early identification and resolution of any significant deficiencies.

(5) The program should efficiently use available resources and minimize requirements for additional resources by NRC or industry.

Thus, the SEP review provides (1) an assessment of the significance of differences between current technical positions on safety issues and those that existed when a particular plant was licensed, (2) a basis for deciding how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety.

The results of the technical evaluations performed under the SEP for the Palisades plant are documented in the SEP Integrated Plant Safety Assessment Report (IPSAR) and the supplement to the IPSAR (NUREG-0820 and NUREG-0820, Supplement 1). In the interest of eliminating unnecessary duplication, issues that were resolved under the SEP will not be discussed here; however, Section 1.4 presents a discussion and update of the status of those issues that were not fully resolved. The remainder of this SER addresses other significant issues not covered under the SEP.

This SER also addresses the status of requirements stemming from the accident at Three Mile Island Unit 2, unresolved safety issues, and other important plant-specific licensing issues that have not yet been resolved.

As noted in this SER, the staff is currently reviewing a number of ongoing licensing actions for Palisades. The staff has determined that these items do not have to be resolved before the FTOL is issued and should not delay the POL to FTOL conversion process. All of these items will be addressed as routine operating reactor licensing actions after the FTOL is issued.

In accordance with the provisions of the National Environmental Policy Act of 1969 (NEPA), the staff prepared a Final Environmental Statement (FES) and an FES supplement (NUREG-0343) that stated in detail the considerations related to plant operation at the design power level. The FES was issued in June 1972, and the supplement to the FES was issued in February 1978. Because they were issued a number of years ago, the staff reexamined the impacts initially examined in these documents. This review, documented in an environmental assessment issued on October 22, 1990, has not noted any significant new environmental impacts or any other significant changes from those identified previously. The staff, accordingly, concluded that another FES supplement is not needed.

1.2 Plant and Plant Site

The Palisades plant is a pressurized-water reactor (PWR) designed by Combustion Engineering, Inc. The licensee, Consumers Power Company, filed the application for a construction permit and operating license in June 1966. The construction permit was issued on March 14, 1967. The initial submittal of the Final Safety Analysis Report was filed on November 5, 1968, and the provisional operating license (DPR-20) was issued on March 24, 1971. A full-power license was issued on October 16, 1972. By letter dated January 22, 1974, the licensee applied for an FTOL. The licensed thermal-power rating currently is 2530 megawatts thermal (MWt).

The Palisades plant is located on the eastern shore of Lake Michigan in Van Buren County, Covert Township, near South Haven, Michigan. The nearest population center to Palisades is the combined twin cities of Benton Harbor and St. Joseph,

approximately 16 miles south of the plant site. The nearest community with a population of 1000 or more is South Haven about 4-1/2 miles north of the site on the shores of Lake Michigan. The population growth in the areas has been modest and the trend is not expected to change.

The site exclusion area is encompassed by the property boundary and is entirely owned and controlled by CPCo. The minimum exclusion distance is 2220 feet (677 meters), and the low population zone (as defined by 10 CFR Part 100) extends to 3 miles from the site. The plant site conforms to current licensing criteria for population distribution.

The primary (reactor) coolant system configuration consists of a reactor pressure vessel (RPV) with two outlet pipes (hot legs) delivering reactor coolant to two U-tube generators which have two return pipes (cold legs) each to the RPV. Each cold leg has a coolant circulating pump. Additionally, a pressurizer is attached to one hot leg. A schematic drawing of the primary coolant system is shown in Figure 1.1.

The secondary (steam) system consists basically of the turbine/generator, the condenser, and the feedwater system. Saturated steam is supplied to the turbine from the steam generator headers, where the steam is expanded through two high-pressure turbines, then flows through four moisture-separator reheaters and intercept valves to two double-flow, low-pressure turbines. The high- and low-pressure turbines are arranged in tandem.

The main feedwater system consists of two condensate pumps driven by electric motors, low-pressure heaters, two feedwater pumps driven by steam turbines, and high-pressure heaters.

The reactor containment structure is a reinforced, post-tensioned concrete dome and cylinder supported on a reinforced-concrete slab. The containment has an internal free volume of 1,640,000 cubic feet. A 1/4-inch-thick welded-steel liner is attached to the inside face of the concrete shell to ensure a high degree of leak-tightness. The cylindrical reinforced-concrete walls are 3-1/2 feet thick, and the concrete dome is 3 feet thick. The thickness of the concrete base slab varies from 8-1/2 to 13 feet. Access is gained during operation by means of a double-door personnel airlock, and a double-gasketed equipment hatch. An emergency escape hatch is also provided.

1.3 Operating History and Experience

The Palisades plant received a provisional operating license in March 1971 and began commercial operation on December 31, 1971. In March 1974, the plant was modified to allow operation with a closed cooling cycle using mechanically operated induced-draft cooling towers. Before that time, the once-through cooling system used water from Lake Michigan as the coolant. In 1977, the allowable primary coolant system pressure was increased to 2100 psia from 1800 psia, and authority was granted to operate the reactor at 2530 MWt, instead of at 2200 MWt.

The original design of the Palisades plant included underwater storage racks for spent fuel that could hold a total of 798 fuel assemblies. This capacity was based on the assumption that other storage facilities would be available to accept spent fuel from commercial power plants. Thus, it would be necessary

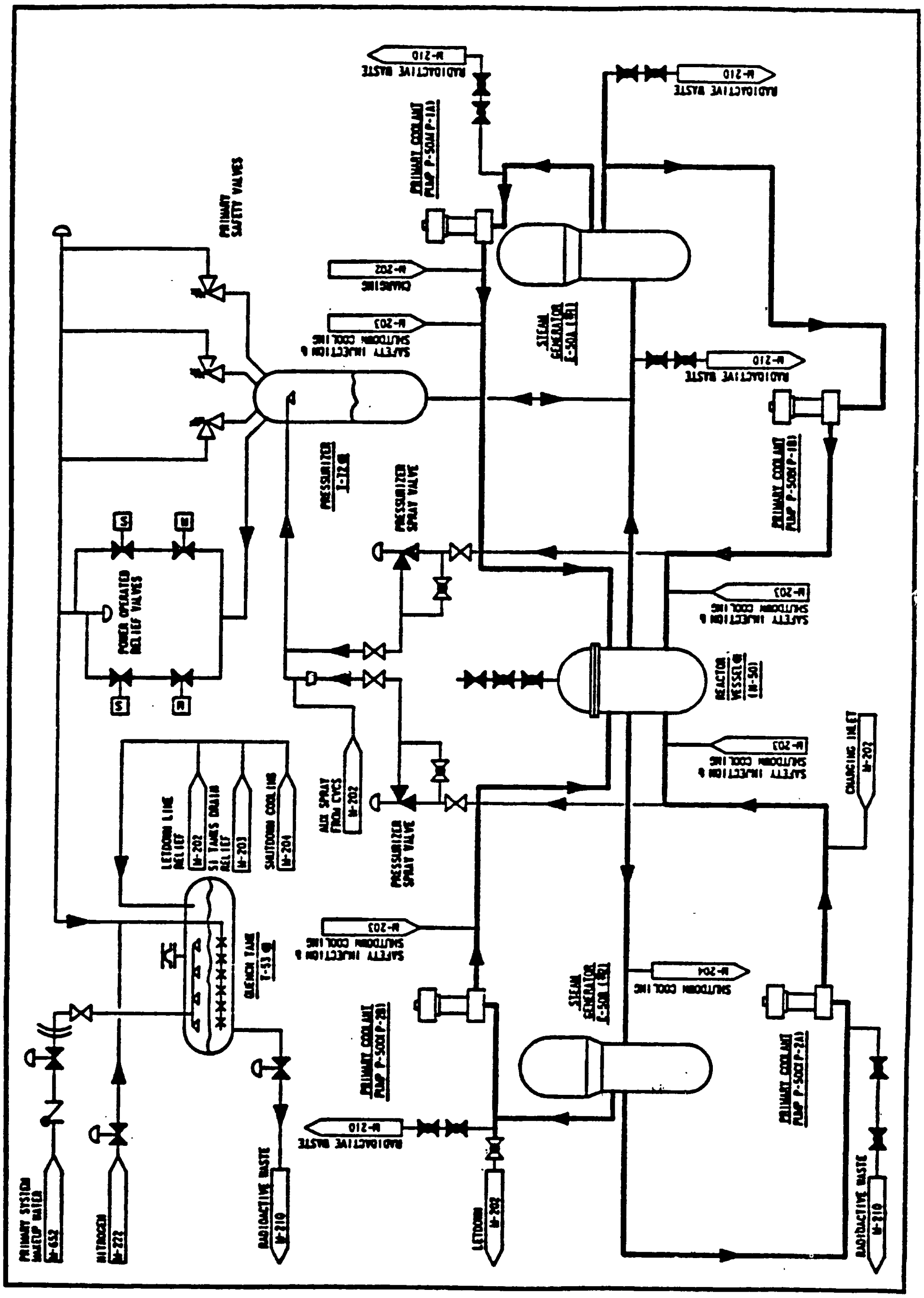

Figure 1.1 Primary coolant system flow

to store spent fuel for only a short period of time before shipping it off site. However, because a change of national policy prohibited reprocessing commercial spent fuel, and because licensed facilities for the ultimate disposal of spent fuel were unavailable, all spent fuel from power operations at all plant sites must be stored on site. Thus, on February 20, 1986, CPCo applied for authority to increase the amount of spent fuel storage at the plant. Approval was granted on July 24, 1987. The present storage capacity of 892 fuel assemblies is sufficient to maintain full-core offload capability until the end of fuel cycle 9. By letter dated August 17, 1990, the licensee notified the NRC that it intends to utilize the general license granted in new Subpart K of 10 CFR Part 72 when it is able to satisfy the conditions of the general license for the onsite storage of spent fuel.

During the early years of operation, coordinated phosphate treatment controlled the pH of secondary water to the steam generator, and sodium sulphite was used to control the oxygen content. By mid-1974, more than 2600 steam generator tubes had been plugged in response to wastage and suspected intergranular corrosion. In 1974, an all-volatile chemistry treatment replaced the use of coordinated phosphate and sodium sulphite. Instead, morpholine and hydrazine were used for pH and oxygen control, respectively.

Tubes continued to leak, and by early 1989 more than 4300 tubes had been plugged in the steam generators, representing approximately 25 percent of the tubes in each generator. In late 1989, CPCo decided to replace both steam generators during the fall of 1990.

The NRC project manager assigned to the FTOL review for the Palisades Nuclear Plant is Mr. Armando Masciantonio. Mr. Masciantonio may be contacted by calling (301) 492-1337 or writing:

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1.4 Systematic Evaluation Program

The Commission initiated the Systematic Evaluation Program to provide a framework for reviewing the designs of older operating nuclear power plants to reconfirm and document their safety. The review provided (1) an assessment of the significance of differences between current technical positions on safety issues and those that existed when a particular plant was licensed, (2) a basis for making decisions on how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety.

The initial review of the Palisades plant as part of the SEP was published in NUREG-0820, the Integrated Plant Safety Assessment Report, dated October 1982. The review compared the as-built plant design with then-current review criteria in 137 different areas defined as "topics." During the SEP review, 47 of the topics were deleted from consideration because (1) the 47 topics were being reviewed under other programs (unresolved safety issues or Three Mile Island Action Plan tasks), (2) the topic was not applicable to the Palisades plant, or (3) the items to be reviewed under that topic did not exist at the site.

Of the original 137 topics, 90 were reviewed for Palisades; of these, 59 met current criteria or were acceptable on another defined basis. The review of the 31 remaining topics found that certain aspects of plant design differed from then-current criteria. These topics were considered in the integrated assessment of the plant, which consisted of evaluating the safety significance as well as other factors of the identified differences from the then-current design to arrive at decisions on whether modification was necessary from an overall plant safety viewpoint. To arrive at these decisions, engineering judgment was used, as were the results of a limited probabilistic risk assessment study.

In general, the staff's positions in the integrated assessment fell into one or more of the following categories:

(1) Equipment modification or addition needed.
(2) Procedure development or Technical Specifications changes needed.
(3) Refined engineering analysis or continuation of ongoing evaluation needed.
(4) No modification needed.

Table 4.1 of the IPSAR summarizes the staff's integrated assessment positions and documents the licensee's agreement with those positions.

For positions classified as either category 1 or 2, Table 4.1 of the IPSAR lists the scheduled completion dates agreed upon by the staff and the licensee. For positions classified as category 3, the licensee gave the staff the results of the ongoing evaluation for review.

The evaluation of these issues and their status is addressed in the IPSAR (NUREG-0820, Supplement 1) published in November 1983. All but three of the issues identified in the IPSAR were closed in Supplement 1. The three remaining issues and their status are discussed below.

(1) <u>SEP Topic III-5A, "Effects of Pipe Breaks Inside Containment"</u>

This issue relates to the potential for a break in the letdown or charging line to result in damage to certain nearby instrument lines (IPSAR Supplement 1, Section 2.3).

CPCo analyzed the piping using the SEP guidelines for mechanistically determining the locations of postulated line breaks. The analysis demonstrated that breaks in the lines in the vicinity of the instrument lines need not be postulated. The staff issued its SER on February 4, 1987, resolving this issue.

(2) <u>SEP Topic III-6, "Seismic Design Issues"</u>

This issue relates to the adequacy of the design of certain structures to withstand seismic motions (IPSAR Supplement 1, Section 2.4). The six open issues were

(i) adequacy of input seismic motions used in the analysis of structures

(ii) amplification of floor response spectra in response to out-of-plane vibration of floors and walls

(iii) analysis of control room panels C-11A and C-126, and switchgear 1D

(iv) justification for qualifying control room panel C-33 on the basis of comparison to motor-control center (MCC) 1 and MCC 2

(v) use of unacceptable vertical accelerations for the analysis of MCC 1 and MCC 2

(vi) development of a plan and schedule to implement generic cable tray evaluation guidelines developed by the SEP owners group

After IPSAR Supplement 1 was issued, CPCo submitted information related to the first, second, third, and sixth issues (above). The staff reviewed the information and issued an SER on October 20, 1986. On the basis of that review, all six of the issues remained unresolved. The sixth issue, relating to cable tray evaluation, will be resolved during implementation of Unresolved Safety Issue (USI) A-46.

CPCo responded to the staff's SER on (1) January 21, 1987, committing to using NUREG/CR-1833 response spectra, and (2) April 30, 1987, addressing the remaining open issues. The staff is reviewing these submittals.

(3) SEP Topic III-7B, "Design Codes and Standards"

After Palisades was licensed, a number of design codes and standards were revised or adopted. If these revised/new codes and standards were to be applied to the Palisades design, in some instances certain designs could be unacceptable. A number of open issues were identified in a staff SER (November 1, 1983). The staff concluded, however, that in no case was any modification required in order to restore a margin of safety; furthermore, the information required was confirmatory in nature and was needed only to complete documentation. A consultant's report prepared for the NRC on the review of CPCo submittals concluded that the only issue not resolved was extreme snow loading on the roof of the spent fuel building. The staff had reviewed the masonry wall issue under the Bulletin 80-11 effort, and the SER dated December 27, 1989, closed it.

A final staff SER has not been issued to reflect the current status of the design code issue.

These remaining SEP issues will be resolved through normal licensing action.

2 SIGNIFICANT OPEN ISSUES

2.1 Three Mile Island Lessons-Learned Requirements

In response to the accident at Three Mile Island Unit 2 (TMI-2), several groups that were established to investigate the accident made recommendations that resulted in NRC requirements. These groups included

- Congress
- General Accounting Office
- President's Commission on the Accident at Three Mile Island
- NRC Special Inquiry Group
- NRC Advisory Committee on Reactor Safeguards
- Lessons Learned Task Force
- Bulletins and Orders Task Force of the NRC Office of Nuclear Reactor Regulation
- Special Review Group of the NRC Office of Inspection and Enforcement
- NRC Siting Task Force
- NRC Emergency Preparedness Task Force
- NRC Office of Standards Development
- NRC Office of Nuclear Regulatory Research.

NUREG-0660, entitled "NRC Action Plan Developed As a Result of the TMI-2 Accident" (referred to as the action plan), was developed to provide a comprehensive and integrated plan for the actions NRC judged necessary to correct or improve the regulation and operation of nuclear facilities. The action plan was based on the experience from the TMI-2 accident and the recommendations of the investigating groups.

With the development of the action plan, NRC transformed the recommendations of the investigating group into discrete scheduled tasks that specify changes in regulatory requirements, organization, or procedures. Some actions to improve the safety of operating plants were judged to be necessary before an action plan could be developed, although they were subsequently included in NUREG-0660. Such actions came from the bulletins and orders issued by the Commission immediately after the accident, the first report of the Lessons Learned Task Force, and the recommendations of the Emergency Preparedness Task Force. Before these immediate actions were applied to operating plants, they were approved by the Commission.

The NRC identified a discrete set of licensing requirements related to TMI-2 in the action plan for Palisades. NUREG-0737, entitled "Clarification of the TMI Action Plan Requirements," was issued in November 1980. This report identified the specific items from NUREG-0660 that were approved by the Commission for implementation at nuclear power plants. It also included additional information about schedules, applicability, method of implementation review, submittal dates, and clarification of technical positions. By Generic Letter 82-33, dated December 17, 1982, the staff issued Supplement 1 to NUREG-0737 to coordinate and indicate initiatives related to

- safety parameter display systems
- detailed control room design reviews
- application of Regulatory Guide 1.97 to emergency response facilities
- upgrading emergency operating procedures (EOPs)
- emergency response facilities
- emergency operations facility (EOF)
- technical support center (TSC)
- operational support center (OSC)
- meteorological data

All of the TMI Action Plan requirements for pressurized-water reactors (PWRs) documented in NUREG-0737 have been resolved for Palisades.

2.2 Unresolved Safety Issues

Unresolved safety issues (USIs) are issues considered on a generic basis after the staff has made the initial determination that the safety significance of the issue does not prohibit continued operation or require licensing actions while the longer term generic review is taking place. All previous staff reviews in these areas are incorporated by reference.

The unresolved safety issues applicable to the Palisades plant for which a response was required are listed below.

USI	Subject
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant System
A-9	Anticipated Transients Without Scram
A-11	Reactor Vessel Material Toughness
A-17	Systems Interactions in Nuclear Power Plants
A-24	Qualification of Class 1E Safety-Related Equipment
A-26	Reactor Vessel Pressure Transient Protection
A-36	Control of Heavy Loads Near Spent Fuel
A-40	Seismic Design Criteria
A-44	Station Blackout
A-46	Seismic Qualification of Equipment in Operating Plants
A-47	Safety Implications of Control Systems
A-49	Pressurized Thermal Shock

Each of these USIs and its status at the Palisades plant are given below.

2.2.1 USI A-2, "Asymmetric Blowdown Loads on Reactor Primary Coolant System"

This unresolved safety issue was resolved in January 1981 with the publication of NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems."

In October 1975, the NRC notified each operating PWR licensee of a potential safety problem concerning the fact that asymmetric loss-of-coolant accident (LOCA) loads had not been considered in the design of any PWR piping system. In June 1976, the NRC informed each PWR licensee that licensees were required to reassess the design of the facility's reactor vessel support. The staff expanded the scope of the problem in January 1978 with a request for additional information to all PWR licensees. NUREG-0609 provided guidance for these analyses. For operating PWRs, Multi-Plant Action (MPA) D-10 was established by NRC's Division of Licensing for implementation purposes.

During the course of the work on USI A-2, it was demonstrated that there were only a very limited number of break locations that could give rise to significant loads. Subsequently, after substantial new technical work, it was demonstrated that pipes would leak before they broke and that new fracture mechanics techniques for the analysis of piping failures assured adequate protection against failures in primary system piping in PWRs (Generic Letter 84-04). This was reflected in (1) a revision to General Design Criterion (GDC) 4 (Appendix A to 10 CFR Part 50) published in the Federal Register in final form on April 11, 1986, and in (2) a subsequent revision to GDC 4 published in the Federal Register on July 23, 1986. Also, it has been satisfactorily demonstrated in the course of the USI A-2 effort that there is very little chance of simultaneous pipe loading with both LOCA and safe-shutdown earthquake (SSE) loads. Therefore, the last revision of GDC 4 represented the final technical action of NRC regarding the asymmetric blowdown loads issue in PWR primary coolant main loop piping.

Implementation and Status

Submittals by Baltimore Gas and Electric Co. on February 15 and June 30, 1980, on behalf of several licensees, including CPCo for the Palisades plant, responded to the January 25, 1978, generic letter requesting evaluation of asymmetric LOCA loads. On February 23, 1981, the staff requested additional generic and plant-specific information; the information was submitted on behalf of CPCo by Baltimore Gas and Electric Co. on July 31, 1981. In a letter dated November 2, 1987, from the Chairman of the Combustion Engineering (CE) owners group to the NRC, documentation of the leak-before-break (LBB) evaluation was provided along with a verification of the applicability to the Palisades plant. The LBB evaluation resolved a specific concern regarding the ability of the fuel assembly support grid to withstand asymmetric LOCA loads.

On October 27, 1989, the staff issued its evaluation of the licensee's submittals. The staff concluded that the licensee gave reasonable evidence that the reactor system at Palisades would withstand the effects of asymmetric LOCA loads and that the reactor could be taken safely to a cold shutdown condition. A question remained regarding the seismic design of the fuel assembly. This unrelated issue will be resolved by a new grid design.

2.2.2 USI A-9, "Anticipated Transients Without Scram"

This unresolved safety issue was resolved in June 1984 with the publication of a final rule (10 CFR 50.62) to require improvements in plants to reduce the likelihood that the reactor protection system (RPS) would fail to shut the reactor down following anticipated transients and to mitigate the consequences of an anticipated transient without scram (ATWS) event.

The rule includes the following design-related requirements:

- 50.62(c)(1) Diverse and independent auxiliary feedwater initiation and turbine trip for all PWRs
- 50.62(c)(2) Diverse scram systems for CE and Babcock & Wilcox (B&W) reactors
- 50.62(c)(3) Alternate rod injection (ARI) for boiling-water reactors (BWRs)

- 50.62(c)(4) Standby liquid control system (SLCS) for BWRs
- 50.62(c)(5) Automatic trip of recirculation pumps under conditions indicative of an ATWS for BWRs

Information requirements and an implementation schedule are also specified.

Implementation and Status

CPCo gave the staff information to demonstrate the adequacy of proposed modifications to comply with the requirements of 10 CFR 50.62(c)(6) by letter dated April 23, 1986. The staff reviewed the document and determined that it needed additional information. CPCo provided the additional information by letter dated June 30, 1987, and also provided an implementation schedule for the proposed modifications. On August 5, 1988, the staff informed CPCo that the shared power supply for the diverse trip system with the RPS was not acceptable. CPCo indicated, by letter dated January 19, 1989, that the design would be changed to provide diverse trip system power supplies separate and independent of the RPS power supplies. On December 5, 1989, the staff issued an SER accepting CPCo's ATWS design. The acceptance is contingent upon a post-implementation audit of Class 1E qualification of certain relays, bistables, and isolation devices. CPCo has scheduled implementation of the ATWS modifications for the end of the next refueling outage--February 1991.

2.2.3 USI A-11, "Reactor Vessel Material Toughness"

This unresolved safety issue was resolved in October 1982 with the publication of NUREG-0744, "Pressure Vessel Material Fracture Toughness." NUREG-0744 was issued by Generic Letter (GL) 82-26 and provided only a methodology to satisfy the requirements of 10 CFR Part 50, Appendix G. Licensees were not required to respond to GL 82-26.

Because of the remote possibility that nuclear reactor pressure vessels designed to the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME) would fail, the design of nuclear facilities does not protect against reactor vessel failure. Prevention of reactor vessel failure depends primarily on maintaining the vessel's material fracture toughness at levels that will resist brittle fracture during plant operation. At service times and under operating conditions typical of current operating plants, fracture toughness properties provide adequate margins of safety against vessel failure; however, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial safety margins.

Appendix G to 10 CFR Part 50 requires that the Charpy upper-shelf energy (USE) throughout the life of the vessel be no less than 50 ft-lb unless it is demonstrated that lower values will provide margins of safety against failure equivalent to those provided by Appendix G to the ASME Code. USI A-11 was initiated to address the staff's concern that some vessels were projected to include beltline materials with Charpy USE less than 50 ft-lb.

NUREG-0744 provides a method for evaluating reactor vessel materials when their Charpy USE is predicted to fall below 50 ft-lb. Plants will use the prescribed method when analysis of irradiation damage predicts that the Charpy USE is below 50 ft-lb. Although NUREG-0744 provides a method for evaluating reactor vessel

materials when their Charpy USE is predicted to fall below 50ft-lb, it does not provide acceptance criteria. Criteria for evaluating material with low Charpy USE is being prepared by the NRC in association with the ASME Code, Section XI, Subgroup on Evaluation Standards. The criteria will require margins of safety against fracture equivalent to those required by Appendix G to 10 CFR Part 50.

Implementation and Status

The licensee has reported that the unirradiated USE in the longitudinal direction for one vessel plate was 120 ft-lb. Using a reduction factor of 0.65, the transverse USE would be 78 ft-lb. Applying Regulatory Guide (RG) 1.99 (Rev. 2), the expected USE at end-of-life would be 43 ft-lb. CPCo has joined the CE owners group to determine the effects of low USE values, that is, less than 50 ft-lb. The staff will be working with the licensee, the owners group, and the ASME Code Subgroup to resolve the issue of low Charpy USE. CPCo is also pursuing an alternate approach, using accelerated irradiation specimens from other plate material along with justification as to the chemical similarity to the limiting plate material. If this approach is satisfactory, the USE will be above 50 ft-lb through end-of-life. The licensee has completed the efforts on the alternate approach and has submitted the results to the NRC for review.

2.2.4 USI A-17, "Systems Interactions in Nuclear Power Plants"

GL 89-18, dated September 6, 1989, was sent to all power reactor licensees and constitutes the resolution of USI A-17. The generic letter did not require any licensee actions.

GL 89-18 had two enclosures which (1) outlined the bases for the resolution of USI A-17 and (2) provided five general lessons learned from the review of the overall systems interaction issue. The staff anticipated that licensees would review this information in other programs, such as the individual plant examination (IPE) for severe accident vulnerabilities. Specifically, the staff expected that insights concerning water intrusion and flooding from internal sources, as described in the appendix to NUREG-1174, would be considered in the IPE program. Also considered in the resolution of this USI was the expectation that licensees would continue to review information on events at operating nuclear power plants in accordance with the requirements of TMI Task Action Plan Item I.C.5 (NUREG-0737).

The NRC sent a letter to CPCo on September 26, 1972, advising the licensee of an event that occurred at the Quad Cities Station in which a failure in the circulating water system caused flooding and degradation of some safety-related equipment. The staff asked CPCo to review the Palisades design to determine if failure of non-Category I equipment could result in a condition, that is, flooding or a release of chemicals, that might impair the performance of safety-related equipment.

Implementation and Status

CPCo responded that the combination of studies done under Systematic Evaluation Program (SEP) Topics III-2, III-4.A, III-5.A, III-5.B, III-6, VII-3, and IX-5, the response to various TMI action plans, and other plans are likely to satisfy a large portion of any criteria the NRC may develop to address systems interactions.

In its October 26, 1972, response to the NRC's letter of September 26, 1972, related to the same issue, CPCo responded that its analyses revealed that the circulating water system has the greatest potential for flooding, and that several areas might be flooded in the event of a break. However, for flooding to affect safety-related equipment, the turbine building (590-foot elevation) would have to be flooded to a depth of more than 3-1/2 feet. This is based on the service water pump motors as the limiting equipment. CPCo concluded that these motors are not likely to experience flooding because water would flow out the doors and through other openings. Furthermore, CPCo reported that all shutdown cooling equipment, component cooling water, emergency diesel generators, and engineered safeguards are protected from flooding by watertight marine doors on all doorways connecting the auxiliary and turbine buildings. Similarly, the auxiliary feed pump room is protected by a watertight door and by seals around penetrations. Spatial isolation and watertight enclosures around redundant safeguards equipment protects against flooding from a local source. Since no chemicals are used in the circulating water and fire protection systems, and water treatment systems are isolated from safety-related systems, the safety-related systems will not be impaired.

2.2.5 USI A-24, "Qualification of Class 1E Safety-Related Equipment"

This unresolved safety issue was resolved in July 1981 with the publication of NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." Part I of the report is the original NUREG-0588 that was issued for comment; that report, in conjunction with the Division of Operating Reactors (DOR) Guidelines, was endorsed by a Commission memorandum and order as the interim position on this subject until "final" positions were established in rulemaking. On January 21, 1983, the Commission amended 10 CFR 50.49 (the rule), effective February 22, 1983, to codify existing qualification methods in national standards, regulatory guides, and certain NRC publications, including NUREG-0588.

The rule is based on the DOR Guidelines and NUREG-0588. These provide guidance on (1) how to establish environmental service conditions, (2) how to select methods considered appropriate for qualifying the equipment in different areas of the plant, and (3) such other areas as margin, aging, and documentation. NUREG-0588 does not address all areas of qualification; it does supplement, in selected areas, the provisions of the 1971 and 1974 versions of the Institute of Electrical and Electronics Engineers (IEEE) Standard 323. The rule recognizes previous qualification efforts completed as a result of Commission Memorandum and Order CLI-80-21 and also reflects other versions of IEEE 323, dependent on the date of the construction permit safety evaluation report. Therefore, plant-specific requirements may vary in accordance with the rule.

In summary, the resolution of USI A-24 is embodied in 10 CFR 50.49. A measure of whether each licensee has implemented its resolution may, therefore, be found in the determination of compliance with 10 CFR 50.49. This was addressed by 72 SERs for operating plants issued shortly after the rule was published and subsequently in operating license reviews pursuant to Standard Review Plan (SRP) Section 3.11 (NUREG-0800). This was addressed further by the first-round environmental qualification inspections conducted by the NRC.

Implementation and Status

The NRC issued an SER on January 31, 1985, which contained the staff's finding that the environmental qualification (EQ) program was in compliance with 10 CFR 50.49. On February 25, 1985, the Commission granted CPCo an extension until November 30, 1985, to meet the requirements on certain specific items which CPCo had identified in a letter dated January 15, 1985.

2.2.6 USI A-26, "Reactor Vessel Pressure Transient Protection"

This unresolved safety issue was resolved in September 1978 with the publication of NUREG-0224, "Reactor Vessel Pressure Transient Protection for PWRs," and SRP Section 5.2. The licensees of all operating PWRs were asked to provide an over-pressure prevention system that could be used whenever the plants were in startup or shutdown conditions. The issue affected all operating and future plants, and the staff established MPA B-04 for implementing the solution at operating PWRs.

Since 1972, there have been numerous reported incidents of pressure transients in PWRs where pressure and temperature limits of the Technical Specifications have been exceeded. The majority of these events occurred while the reactors were in a solid-water condition, during startup or shutdown, and at relatively low reactor vessel temperatures. Since the reactor vessels are less tough at lower temperatures, they are more susceptible to brittle fracture under these conditions than at normal operating temperatures. In light of the frequency of the reported transients and the associated potential for vessel damage, the NRC staff concluded that measures should be taken to minimize the number of future transients and reduce their severity.

GL 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," was published on July 12, 1988. This generic letter provides guidance regarding review of pressure-temperature limits and indicates that licensees may have to revise low-temperature overpressure-protection setpoints.

Implementation and Status

In response to the August 11, 1976, letter related to low-temperature overpressure-protection (LTOP) events, CPCo applied for a license amendment to incorporate technical specifications (TS) related to design features to mitigate LTOP. The staff issued the amendment on September 10, 1979.

The licensee submitted an application to modify the LTOP TS on November 6, 1987, along with supplemental information and revisions on December 22, 1987, and January 20, April 12, and November 1, 1988. The LTOP TS were amended on November 14, 1988. On September 12, 1989, the licensee proposed a TS amendment that would allow the use of a variable setpoint for LTOP using the PORVs. Associated with this proposed technical specification, the licensee also proposed revised Appendix G curves in accordance with GL 88-11 and RG 1.99 (Rev. 2). The staff has reviewed these proposed changes and issued an SER and license amendment dated April 26, 1990.

2.2.7 USI A-36, "Control of Heavy Loads Near Spent Fuel"

This unresolved safety issue was resolved in July 1980 with the publication of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and SRP Section

9.1.5. The staff established MPAs C-10 and C-15 for the implementation of Phases I and II, respectively, of the resolution of this issue at operating plants.

In nuclear power plants, heavy loads may be handled in several plant areas. If these loads were to drop in certain locations in the plant, they may impact spent fuel, fuel in the core, or equipment that may be required to achieve safe shutdown and to continue decay heat removal. USI A-36 was established to systematically examine staff licensing criteria and the adequacy of measures in effect at operating plants, and to recommend necessary changes to ensure the safe handling of heavy loads. The guidelines proposed in NUREG-0612 include definition of safe load paths, use of load handling procedures, training of crane operators, guidelines on slings and special lifting devices, periodic inspection and maintenance for the crane, as well as various alternatives.

By generic letters dated December 22, 1980, and February 3, 1981 (GL 81-07), all utilities were asked to evaluate their plants against the guidance of NUREG-0612 and to provide their submittals in two parts: Phase I (6-month response) and Phase II (9-month response). Phase I responses were to address Section 5.1.1 of NUREG-0612 which covered the following areas:

- definition of safe load paths
- development of load handling procedures
- periodic inspection and testing of cranes
- qualifications, training, and specified conduct of operators
- special lifting devices satisfying the guidelines of American National Standards Institute (ANSI) Standard N14.6.6
- lifting devices, not specially designed, installed and used in accordance with the guidelines of ANSI Standard B30.9
- cranes designed to ANSI Standard B30.2 or Crane Manufacturers Association of America (CMAA) Standard 70

Phase II responses were to address Sections 5.1.2 through 5.1.6 of NUREG-0612 which covered the need for electrical interlocks or mechanical stops, or alternatively, single-failure-proof cranes or load drop analyses in the spent fuel pool area (PWR), containment building (PWR), reactor building (BWR), other areas, and the specific guidelines for single-failure-proof handling systems.

As stated in GL 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants'--NUREG-0612," all licensees have completed the requirement to perform a review and submit a Phase I and a Phase II report. On the basis of the improvements in handling heavy loads obtained from the implementation of NUREG-0612 (Phase I), further action was not required to reduce the risks associated with the handling of heavy loads. Therefore, a detailed Phase II review of heavy loads was not necessary, and Phase II was considered completed.

Although not a requirement, NRC encouraged the implementation of any actions identified in Phase II that were considered appropriate regarding the handling of heavy loads.

Implementation and Status

CPCo responded to Phase I of this unresolved safety issue in submittals between May 1981 and August 1983. The SER issued on November 9, 1983, stated acceptance of CPCo's Phase I submittals. On the basis of its review of the Phase I responses, the staff concluded that implementation of Phase I provided protection so that the risk associated with potential heavy-load-drops was acceptably small and no further action was required.

2.2.8 USI A-40, "Seismic Design Criteria"

The staff has resolved USI A-40 as documented in (1) NUREG/CR-5347, "Recommendations for Resolution of Public Comments on USI A-40," issued in June 1989, and (2) NUREG-1233, "Regulatory Analysis for USI A-40," issued in September 1989.

For plants not covered within the scope of USI A-46, "Seismic Qualification of Equipment in Operating Plants," the staff concluded that tanks in plants that were subject to licensing review by the staff after 1984 had been reviewed to current requirements and found acceptable. For tanks in plants reviewed between 1980 and 1984, the staff identified four plant sites (six units) that were not explicitly reviewed to current requirements. The four plants (Callaway, Wolf Creek, Shearon Harris, and Watts Bar) are being handled on a plant-specific basis.

USI A-40 originated in 1977. The basic objectives were to (1) study the seismic design criteria, (2) quantify the conservatism associated with the criteria, and (3) recommend modifications to the SRP if changes are justified. Lawrence Livermore National Laboratory (LLNL) completed the study and published its findings in NUREG/CR-1161, "Recommended Revisions to USNRC--Seismic Design Criteria," in May 1980. The report recommended specific changes to the SRP. NRC staff reviewed the report and developed some other changes that would reflect the then-current state of seismic design practices. The resulting SRP changes were issued for public comment in June 1988, and the final SRP changes were published.

The major SRP changes consist of (1) clarification of development of site-specific spectra, (2) justification for use of single synthetic time-history by power spectral density function, (3) location and reductions of input ground motion for soil-structure interaction, and (4) design of above-ground vertical tanks. Except for item 4, these changes do not constitute any additional requirements for current licenses and applications, and thus, no backfitting is being required for these items. However, the revised provisions could be used for margin studies and reevaluations or individual plant examinations for external events.

The utilities participating in the Seismic Qualification Utility Group (SQUG) agreed to implement the changed criteria for flexible vertical tanks for their plants. The staff sent a request-for-information letter (10 CFR 50.54(f)) to the individual utilities operating the four plants. If the responses indicate that large above-ground vertical tanks do not meet the new criteria, plant-specific backfits will be considered.

Implementation and Status

CPCo, as licensee for the Palisades plant, is a participant in the SQUG; therefore, resolution will be achieved in conjunction with SQUG activities.

2.2.9 USI A-44, "Station Blackout"

This unresolved safety issue was resolved in June 1988 with the publication of a new rule (10 CFR 50.63) and RG 1.155.

Station blackout means the loss of offsite ac power to the essential and nonessential electrical buses concurrent with turbine trip and the unavailability of the redundant onsite emergency ac power systems. WASH-1400 (redesignated NUREG-75/014) showed that station blackout could be an important contributor to risk, and operating experience has indicated that ac power systems might be less reliable than originally anticipated. For these reasons, station blackout was designated as an unresolved safety issue in 1980. A proposed rule was published for comment on March 21, 1986. A final rule, 10 CFR 50.63, was published on June 21, 1988, and became effective on July 21, 1988. RG 1.155 was issued at the same time as the rule and references an industry guidance document issued by the Nuclear Management and Resources Council (NUMARC 87-00). In order to comply with the resolution of USI A-44, licensees will be required to

- Maintain onsite emergency ac power supply reliability above a minimum level.
- Develop procedures and training for recovery from a station blackout.
- Determine how long a station blackout the plant should be able to withstand.
- Use an alternate qualified ac power source, if available, to cope with a station blackout.
- Evaluate the plant's actual capability to withstand and recover from a station blackout.
- Backfit hardware modifications if necessary to improve coping ability.

Section 50.63(c)(1) of the rule required each licensee to submit a response, including the results of a coping analysis, within 270 days from issuance of an operating license, or the effective date of the rule, whichever is later.

Implementation and Status

As required by the rule, CPCo responded by letter on April 17, 1989. CPCo has evaluated Palisades using NUMARC 87-00 guidance, except where RG 1.155 has precedence. The duration of station blackout is estimated to be 4 hours, and the alternate ac power source has been selected. The selected option is to improve the reliability of the offsite power supply. Some modifications were made during the 1988 refueling outage and the fall 1989 maintenance outage. Final modifications will be completed by the end of the 1990 refueling outage. The staff is reviewing the CPCo response and will issue an SER.

2.2.10 USI A-46, "Seismic Qualification of Equipment in Operating Plants"

USI A-46 was resolved with the issuance of GL 87-02 on February 19, 1987, which endorsed the approach of using the seismic and test experience data proposed by the Seismic Qualification Utility Group (SQUG) and the Electric Power Research Institute. This approach was endorsed by the Senior Seismic Review and Advisory Panel and approved by the NRC staff.

The scope of the review was narrowed to equipment required to bring each affected plant to hot shutdown and maintain it there for a minimum of 72 hours. The review requires a walkthrough of each plant to inspect equipment. Evaluation of equipment will include (1) adequacy of equipment anchorage, (2) functional capability of essential relays, (3) outliers and deficiencies (i.e., equipment with non-standard configurations), and (4) seismic systems interaction.

As an outgrowth of the Systematic Evaluation Program, the need was identified for reassessing design criteria and methods for the seismic qualification of mechanical and electrical equipment. Therefore, the seismic qualification of the equipment in operating plants must be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this issue was to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at operating plants in lieu of attempting to backfit current design criteria for new plants.

GL 87-02, with associated guidance, required all affected utilities to evaluate the seismic adequacy of their plants. The specific requirements and approach for implementation are being developed jointly by the SQUG and the staff on a generic basis before individual member utilities proceed with plant-specific implementation.

Implementation and Status

The SQUG submitted the Generic Implementation Procedure (GIP), Revision 0, on June 3, 1988. The staff issued a generic safety evaluation (SE) on July 29, 1988, endorsing much of the GIP but with approximately 70 open items requiring resolution. After a series of meetings, SQUG submitted Revision 1 to the GIP on December 23, 1988. The staff issued a supplemental SE in June 1990. Revision 2 of the GIP was submitted September 21, 1990. The staff will issue another SE after it reviews Revision 2 of the GIP. Implementation closeout is projected for 1994.

For Palisades, CPCo committed to a seismic verification plant walkdown, as required by the GIP, by the end of the second refueling outage after the SQUG and NRC resolve the open issues and the final SER is issued.

2.2.11 USI A-47, "Safety Implications of Control Systems"

USI A-47 was resolved on September 20, 1989, with the publication of GL 89-19.

The generic letter states:

> The staff has concluded that all PWR plants should provide automatic steam generator overfill protection, all BWR plants should provide

automatic reactor vessel overfill protection, and that plant procedures and Technical Specifications for all plants should include provisions to verify periodically the operability of the overfill protection and to assure that automatic overfill protection is available to mitigate main feedwater overfeed events during reactor power operation. Also, the system design and setpoints should be selected with the objective of minimizing inadvertent trips of the main feedwater system during plant startup, normal operation, and protection system surveillance. The Technical Specifications recommendations are consistent with the criteria and the risk considerations of the Commission Interim Policy Statement on Technical Specifications Improvement.

In addition, the staff recommends that all BWR recipients reassess and modify, if needed, their operating procedures and operator training to assure that the operators can mitigate reactor vessel overfill events that may occur via the condensate booster pumps during reduced system pressure operation.

Also, page 2 of the generic letter gives additional actions for CE and B&W plants. The generic letter amplifies guidance for licensees.

The generic letter requires that licensees send NRC their schedules and commitments within 180 days of the letter's date. Actions on which commitments are made should be scheduled for implementation before startup after the first refueling outage, but no later than the second refueling outage, beginning 9 months after receipt of the letter.

Implementation and Status

GL 89-19 was issued on September 20, 1989. CPCo responded as part of a CE group on March 20, 1990, and concluded that the recommendations should not be implemented at Palisades at this time but will be addressed under the IPE program. The CPCo response is being reviewed by the staff.

2.2.12 USI A-49, "Pressurized Thermal Shock

The Commission approved the final rule (10 CFR 50.61) on pressurized thermal shock (PTS) in July 1985. RG 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for PWRs," was published in February 1987. Thus, this issue was resolved and new requirements were established, applicable to PWRs only. The rule required that each operating reactor meet the screening criteria given in the rule or provide supplemental analysis to demonstrate that PTS is not a concern for the facility.

Neutron irradiation of reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. The fracture toughness sensitivity to radiation-induced change is increased by the presence of certain materials such as copper. Decreased fracture toughness makes it more likely that, if a severe overcooling event occurs, followed by or concurrent with high vessel pressure, and if a small crack is present on the vessel's inner surface, that crack could grow to a size that might threaten vessel integrity.

Severe pressurized overcooling events are improbable since they require multiple failures and improper operator performance. However, the occurrence of certain

precursor events could potentially threaten vessel integrity if additional failures occurred and/or if the vessel had been more highly irradiated. Therefore, the possibility of vessel failure due to a severe pressurized overcooling event cannot be ruled out.

Implementation and Status

By letter dated November 30, 1988, CPCo submitted information on its fluence reduction efforts to comply with 10 CFR 50.61. This information was updated on April 3, 1989, and May 17, 1990. In this latest submittal, CPCo concluded that the PTS criteria will be exceeded at the axial welds in September 2001. Therefore, the flux reduction achieved to date is insufficient to allow plant operation to the end of the current nominal license term of 2007.

However, CPCo is also pursuing other measures to extend vessel lifetime, such as: greater flux reductions, analysis following RG 1.154, vessel shielding, and so forth, to allow plant operation to the nominal end of plant life and beyond.

2.3 Plant-Specific Licensing Issues

The staff is presently reviewing a number of licensing actions that are unique to the Palisades plant. These are

- steam generator replacement
- transfer of plant ownership
- reactor vessel embrittlement
- station blackout
- restructured Standard Technical Specifications
- embrittlement of reactor vessel supports

The status of each of these items is summarized in the pages that follow.

2.3.1 Steam Generator Replacement

During the operating history of the Palisades plant, the steam generators have been afflicted by a number of corrosion-related problems associated with wastage, intergranular attack, and denting. As a consequence of these problems, a substantial portion of the excess heat transfer capacity of the steam generators has been removed. With the uncertainty that exists with regard to future plugging requirements and the expected useful life of the existing steam generators, CPCo has decided that complete replacement of the steam generators is the preferred method of repair.

The steam generators will be removed from the containment and will be replaced through a 27-foot by 30-foot opening to be cut in the southeast containment wall. The necessary rigging and site preparation will precede the actual removal and replacement.

The replacement steam generators are designed to match the essential physical boundaries of the existing steam generators and be compatible with the performance characteristics used in the Final Safety Analysis Report and the license for operation at 2530 MWt, even though the replacement steam generators are designed for operation at 2650 MWt.

Both new steam generators have been procured and have been delivered to the Palisades site. There are no unique engineering or construction aspects to the removal and replacement program. Conventional nuclear industry manufacturing and construction methods will be used. The temporary opening in the containment will be closed in a manner similar to that used to close the original opening. The transport and rigging of the steam generators will use proven techniques. The repair program will rely on fabrication and construction practices and techniques which have been previously qualified for similar applications. The old steam generators will be stored at the Palisades site.

The staff is reviewing issues related to the steam generator replacement as a normal licensing action.

2.3.2 Transfer of Plant Ownership

By letter dated February 27, 1989, CPCo submitted a license amendment request to transfer ownership of the Palisades plant from CPCo to a newly formed Palisades Generating Company. The new company would be formed by a joint venture consisting of CPCo (44% ownership), Bechtel (33% ownership), and Westinghouse (23% ownership).

Palisades Generating Company would consist of 8 to 12 persons, 7 of whom are on the Board of Directors (3 from CPCo, 3 from Bechtel, 1 from Westinghouse).

The operating agreement would retain CPCo as the plant operator for the first five years; no changes are contemplated in the operational staff. Other government agencies that have review and approval authority have been notified.

The staff is reviewing the license amendment request. The results will be provided at a later time as part of normal licensing procedures.

2.3.3 Reactor Vessel Embrittlement

Resistance to brittle fracture, a rapidly propagating catastrophic failure mode for a component containing flaws, is described quantitatively by a material property generally denoted as "fracture toughness." Fracture toughness has different values and characteristics depending on the material under consideration. For steels used in nuclear reactor pressure vessels, three considerations are important: (1) fracture toughness increases with increasing temperature, (2) fracture toughness decreases with increasing load rates, and (3) fracture toughness decreases with neutron irradiation. In recognition of these considerations, power reactors are operated within restrictions imposed by the Technical Specifications on the pressure during heatup and cooldown operations. These restrictions ensure that the reactor vessel will not be subjected to a combination of pressure and temperature that could cause brittle fracture if there were significant flaws in the vessel materials. The effect of neutron radiation on the fracture toughness of the vessel material is accounted for in developing and revising these Technical Specifications limitations.

For the service time and operating conditions typical of current operating plants, reactor vessel fracture toughness for most plants provides adequate margins of safety against vessel failure under operating, testing, maintenance, and anticipated transient and accident conditions over the life of the plant. The principal objective of Task A-11 was to develop an improved engineering

method and safety criteria to allow a more precise assessment of the safety margins during normal operations and transients in older reactor vessels with marginal fracture toughness and of the safety margins during accident conditions for all plants. Requirements for demonstrating vessel toughness margins are given in NUREG-0744, Revision 1, "Resolution of Reactor Vessel Materials Toughness Safety Issue," transmitted by GL 82-26, "Pressure Vessel Material Fracture Toughness."

Appendices G and H to 10 CFR Part 50 require that compliance with minimum fracture toughness requirements be demonstrated and that a materials surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region be maintained.

This issue was discussed during the review of SEP Topic V-6, "Reactor Vessel Integrity," in NUREG-0569, "Evaluation of the Integrity of SEP Reactor Vessels." Resolution of the SEP topic is reported in the Palisades IPSAR, NUREG-0820, Section 3.2. Subsequently, the staff issued GL 88-11, "NRC Position on Radiation Embrittlement and Its Impact on Plant Operations." This letter transmitted a copy of RG 1.99, Revision 2, and asked licensees to predict the effect of neutron radiation on reactor vessel material as required by 10 CFR Part 50, Appendix G, Paragraph V.A, using the methods described in RG 1.99, Revision 2.

By letter dated September 12, 1989, the licensee proposed to revise the pressure-temperature operating limits in Section 3.1 of the Palisades Technical Specifications. The licensee stated that the revised curves were generated in accordance with the methods defined in RG 1.99, Revision 2, and the stress allowed by 10 CFR Part 50, Appendix G, will not be exceeded.

The staff has reviewed the licensee's response to GL 88-11. The results of the staff's review were provided in an SER dated April 26, 1990.

2.3.4 Station Blackout

Electrical power for safety systems at nuclear power plants must be supplied by at least two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet these requirements. Each electrical division for safety systems includes two offsite ac power connections, a standby emergency diesel generator (EDG) ac power supply, and dc sources.

Generic Task A-44 involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all ac power; that is, a loss of both the offsite and the EDG power supplies. This issue arose because of operating experience regarding the reliability of ac power supplies. There have been numerous reports of EDGs failing to start and run in operating plants during periodic surveillance tests. In addition, a number of operating plants have experienced a total loss of offsite electrical power, and more total losses are expected to occur in the future. In almost every one of the loss-of-offsite-power events, the onsite emergency ac power supplies were available immediately to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies has been unavailable. In a few cases, there has been a complete loss of ac power, but during the events, ac power was restored in a short time without serious consequences.

A loss of offsite power involves a loss of both the preferred and backup sources of offsite power. If all offsite power is lost, the onsite emergency ac power system will provide ac power to safety-related equipment.

With respect to emergency onsite ac power, the Palisades design uses two independent engine-driven EDGs of equal size. The generators are designed to provide separate, dependable, onsite power sources capable of starting and supplying essential loads to shut down the plant and maintain it in a safe shutdown condition. The staff evaluated these systems within the framework of SEP Topic VIII-2 and found them acceptable. The staff's evaluation is presented in the Palisades IPSAR, NUREG-0820, Section 3.3.3.

On June 21, 1988, the Commission finalized the station blackout rule, 10 CFR 50.63, which resolves and supersedes Generic Task A-44. The station blackout rule is implemented by Multiplant Action (MPA) A-22. Compliance with this MPA item will be achieved through normal licensing action.

In its most recent action to address MPA A-22, the licensee submitted a response to the station blackout rule by letter dated April 17, 1989. The staff is reviewing this response.

2.3.5 Restructured Standard Technical Specifications

CPCo, as a member of the Combustion Engineering (CE) owners group, has indicated that it will adopt CE Standard Technical Specifications (STS) for the Palisades plant. The staff is reviewing the CE STS. The licensee will submit the Palisades restructured STS after the staff gives the CE STS final approval.

The staff will review the submittal as part of the normal licensing process.

2.3.6 Embrittlement of Reactor Vessel Supports

Generic Safety Issue (GSI) 15, "Radiation Effects on Reactor Vessel Supports," addressed the potential problem of radiation embrittlement of reactor pressure vessel (RPV) support structures. This issue was originally identified as a candidate unresolved safety issue in 1981 and was assigned a priority ranking of "low" in 1983. However, on the basis of data and analyses developed by the Oak Ridge National Laboratory in April 1988, the NRC staff concluded that the potential for RPV support embrittlement from neutron radiation damage could be greater than predictions based on pre-1988 data (NUREG/CR-5556).

After a reevaluation of the issue, it was concluded in December 1988 that the issue should be given a "high" priority ranking. Although the more recent ORNL radiation data suggest that a potential problem may exist for RPV supports, preliminary analysis by the staff led it to conclude that this problem does not pose an immediate concern to public safety. At the same time, reasons exist for resolving this issue. Further investigation is needed to assess the short-term and long-term radiation effects on RPV supports exposed to low-temperature, low-flux radiation. The problem is complicated by uncertainties about the chemical composition and mechanical properties of the reactor support structure both preceding and resulting from radiation.

The staff is reviewing a draft task action plan that describes the scope, the schedule, and the proposed program for resolving GSI-15. A major goal of the

proposed program will be to determine the time frame of the problem; that is, whether there is a correction which should be implemented in the near term or should be relegated to considerations of plant aging and license extension. As part of the resolution of this issue, the licensee for Palisades will be asked to assess the effect of neutron irradiation embrittlement on the integrity of the plant's reactor vessel supports.

3 EXEMPTIONS FROM THE CODE OF FEDERAL REGULATIONS

The licensee was asked to review its records to identify the exemption from the Code of Federal Regulations under which Palisades is presently operating. By letter dated August 17, 1990, the licensee stated that Palisades is presently operating under one exemption from 10 CFR Part 50, Appendix J, and four exemptions from the requirements of 10 CFR Part 50, Appendix R. The exemptions were granted in response to CPCo requests, subsequent to staff review, and will remain in effect under the Palisades FTOL. The exemptions are as follows:

3.1 Exemption From 10 CFR 50, Appendix J

This exemption was approved by NRC in a safety evaluation report (SER) dated December 6, 1989. The exemption provides partial relief from the requirement to test the containment airlocks at or above the calculated design-basis accident peak containment pressure and permits the substitution of a between-the-seals leak test at reduced pressure.

3.2. Exemptions From 10 CFR 50, Appendix R

1. Section III.G.3--Engineered Safeguards Panel Room

   Exemption to requirement for fire detection and fixed fire suppression

   Original CPCo submittal: July 25, 1983
   Revised: July 16, 1984
   Approved by NRC SER: July 12, 1985

2. Section III.G.3--Corridor Between Charging Pump Room and Switchgear Room 1C

   Exemption to requirement for fire detection and fixed fire suppression

   Original CPCo submittal: July 25, 1983
   Revised: July 16, 1984
   Approved by NRC SER: July 12, 1985

3. Section III.G.3--Control Room

   Exemption to requirement for fire detection and fixed fire suppression

   Original CPCo submittal: July 1, 1982
   Approved by NRC SER: February 8, 1983

4. Section III.G.2--Cable Separation Inside Containment

   Exemption to requirement concerning intervening combustibles installed between redundant instrumentation cable trays

Original CPCo submittal: July 20, 1984
Additional information submitted: December 28, 1984
Approved by NRC SER: July 23, 1985

5. One schedular exemption request to Appendix R is still awaiting NRC approval.

4 CONCLUSIONS

On the basis of its evaluation of the application as detailed in the preceding sections, the staff has determined the following:

(1) The application for a full-term operating license (FTOL) for the Palisades plant filed by Consumers Power Company, dated January 22, 1974, complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations in 10 CFR Chapter I, except as duly exempted therefrom.

(2) The provisions of Provisional Operating License DPR-20 have been met.

(3) The facility will operate in conformity with the FTOL application, the provisions of the Act, and the rules and regulations of the Commission.

(4) There is reasonable assurance (a) that the activities authorized by the FTOL can be conducted without endangering the health and safety of the public and (b) that such activities will be conducted in compliance with the regulations of the Commission as given in 10 CFR Chapter I.

(5) The licensee is technically qualified to engage in the activities authorized by the FTOL in accordance with the regulations of the Commission as given in 10 CFR Chapter I.

(6) The issuance of the FTOL will not be inimical to the common defense and security or to the health and safety of the public.

(7) The NRC should authorize the FTOL for the Palisades plant.

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11. ABSTRACT *(200 words or less)*

This safety evaluation report relates to the issuance of a full-term operating license (FTOL) for the Palisades Nuclear Generating Plant in response to an application filed by the Consumers Power Company. The report provides an update of the status of the Systematic Evaluation Program (SEP) issues that were not fully resolved in the SEP Integrated Plant Safety Assessment Report (NUREG-0820 and NUREG-0820 Supplement 1). This report also addresses the status of requirements stemming from the accident at Three Mile Island Unit 2, unresolved safety issues, and other important plant-specific issues that have not yet been resolved.

The staff has evaluated the issues related to the conversion of the provisional operating license to a full-term operating license and concluded that the facility can continue to be operated without endangering the health and safety of the public.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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