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February 28, 1985

James G Keppler, Administrator
Region III
US Nuclear Regulatory Commission
799 Roosevelt Road
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DOCKET 50-255 - LICENSE DPR-20 -
PALISADES PLANT - 1984 10CFR50.59 ANNUAL REPORT OF FACILITY CHANGES, TESTS AND EXPERIMENTS

Attached is Consumers Power Company's 10CFR50.59 Annual Report describing the Facility Changes (FC), Specification Field Changes (SFC), tests and experiments performed in 1984 at the Palisades Plant which are reportable under 10CFR50.59(b).

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Attachment

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ATTACHMENT

CONSUMERS POWER COMPANY

Palisades Plant - Docket 50-255

1984 10CFR50.59 ANNUAL REPORT OF

FACILITY CHANGES, TESTS AND EXPERIMENTS

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12 Pages

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PALISADES PLANT

FC-375

This modification covered the installation of high density storage racks in the spent fuel pool.

Safety Evaluation Summary

The new design of the spent fuel storage racks was a requirement to optimize fuel storage capability. Consideration was given to heavy object handling over the spent fuel pool. Installation was done according to approved plant procedures. The modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment previously evaluated in the FSAR.

FC-407-14A

The modification under this phase of the fire protection modifications covers the installation of the auxiliary hot shutdown panel C150.

Safety Evaluation Summary

The modification does not affect the design, operation, or function of any equipment important to safety. It is the intent that this modification will provide additional reliability of existing systems important to safe shutdown of the plant in case of fire in the Control Room.

FC-407-14E

This modification was part of the fire protection modification. This particular phase relocated the power source for the diesel generator crank case exhauster motors of diesel generators 1-1 and 1-2 from MCC 1 and 2, respectively to the generator bus of each respective diesel generator. It also identified, by unique markings, external wires to buses 1-B and 1-C such that local control of the diesel generators could be established in case of fire.

Safety Evaluation Summary

The crank case exhausters operate only when the diesel generators are operating and are required for safe, reliable operation of the diesel generators. A cable spreading room fire presents the possibility of separation of off-site power and destruction of the 480V power sources in the room, thus rendering exhausters inoperable. Relocating the power source for the crank case exhauster motors to the generator bus of the diesel generator provides a reliable source to supply the exhausters.

The above postulated fire also presents the possibility of damage to control circuitry from the control room to the diesel generators and the 2400V buses 1C and 1D. Marking the terminals when this circuitry enters the buses will

FC-407-14E - Continued

expedite disconnecting the damaged circuitry and, thus, assure local control and operation. Therefore, the possibility or the consequences of an accident or malfunction of equipment previously evaluated in the FSAR is not increased.

FC-450

This modification added two snubbers to each of the B and D SI bottle lines to reduce nozzle stresses during an earthquake.

Safety Evaluation Summary

The addition of these snubbers corrected a design error in the piping system. The design now meets original specifications for piping stresses and reduces the probability of occurrence or consequence of an accident or malfunction of equipment important to safety. It assures SI water will be able to get to the core and lessens the likelihood of a small break LOCA at the nozzles.

FC-452-2

This Facility Change revised the CCW flow containment isolation logic from SIS only, to ISI and CCW low pressure.

Safety Evaluation Summary

The resulting increase in PCP availability provided by the above reduces the probability of occurrence or consequences of an accident or malfunction of equipment important to safety.

The additional heat load provided by PCP's will not overload the CCW system. Other heat loads are considered negligible. The low pressure input to the isolation logic will insure adequate flow to the shutdown cooling heat exchanger if system pressure drops below the auto start set point for the standby CCW pump, assuming one pump is operating.

FC-465

Five mechanical snubbers were installed on safety related piping under this modification. Two were on the FW inlet piping to the steam generators, one was on the FW tie line on the FW inlet, and one was on an SI bottle line. This phase of the snubber modifications was in addition to FC-450 to correct a design error in the piping system.

Safety Evaluation Summary

The snubbers were installed to improve the performance of the piping involved during a seismic event. The piping system is now stiffer under the new dynamic conditions. The dynamic response as analyzed resulted in identifying

FC-465 - Continued

stresses to be within FSAR and ANSI B31.1 1969 allowable values. As these snubbers are passive under static and thermal transient loads, no new mode of failure is introduced and the possibility of an accident or malfunction of a different type is not present.

FC-482

This modification covered the installation of redundant level transmitter, power supply, and indicating alarm on the condensate storage tank, and the updating of the existing level transmitter.

Safety Evaluation Summary

This modification provided redundant level indication and updates the existing level transmitter. Therefore, it decreases the probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR.

FC-494-1

This modification covered the installation of two high range Gamma radiation monitors in containment, including all necessary cabling and conduit from the monitors to new control room panel C-11A.

Safety Evaluation Summary

The equipment installed provides reliable radiation monitoring. The equipment, itself, does not provide any automatic control action and response is through operator action. Therefore, the possibility of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased.

FC-494-2

This Facility Change consisted of four parts:

1. Removing an existing auxiliary building equipment gas analyzer panel and locating selected sample points.

2. Installing a new post accident sampling panel (C103-1) to be used for sampling primary and other sample points before and after an accident.

3. Addition of necessary sampling lines for new post accident sampling panel.

4. Addition of necessary electrical changes required for the projects.

FC-494-2 - Continued

Safety Evaluation Summary

This equipment is being installed to provide reliable post accident sampling equipment to improve operator action in mitigating a design based accident. This equipment does not provide any automatic control action. This equipment is being installed in accordance with all codes and standards required for its use in a nuclear facility. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased.

FC-494-4

The following modifications were covered by this Facility Change:

1. The installation of two new hydrogen monitoring panels in auxiliary building at separate locations.

2. The installation of remote monitor panels in Control Room on C-11A panels.

3. The installation of necessary sample lines inside containment.

4. Two piping penetrations were changed to meet the needs of the two hydrogen monitor panels.

Safety Evaluation Summary

This equipment is being installed to provide highly reliable hydrogen monitoring to be used in mitigating a design based accident. The equipment does not provide any automatic control action and is qualified and installed to all required codes and standards for use in a nuclear facility. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased.

FC-494-5

Two separate main steam line radioactive monitors were installed in the auxiliary building under this modification, along with remote readout panels in Control Room panel C-11A.

Safety Evaluation Summary

The equipment installed provides reliable radiation monitoring to be used for information. The equipment does not provide any automatic control action. Therefore, the probability of an accident or malfunction important to safety as previously described in the FSAR is not increased.

FC-519

This modification covered the structural upgrading of block walls near safety related equipment which were generally proven deficient as a result of seismic analysis, in response to IE Bulletin 80-011.

Safety Evaluation Summary

Block walls were upgraded near safety related equipment. Therefore, the probability of safety related equipment failure is decreased. The failure mode of a block wall relates only to its structural capability, thus there is no possibility of creating a different type of accident or malfunction condition.

FC-558

This Facility Change covered the electrical system extension, the function of which increases the 480V distribution capacity of the Class IE power system. The additional capacity will supply power to the new Class IE HVAC system, ESF sump pumps, reconnect CC #1 and #2 from LC #11 and #12, and future loads requiring Class IE power.

Safety Evaluation Summary

This modification decreases the probability of occurrence or consequences of an accident important to safety by providing a greater degree of reliability to the Class IE 480V power system, and will provide increased capacity for future loads requiring Class IE power. The equipment for this modification is qualified for use in nuclear power plants' safety related systems.

FC-562

This modification provided load shedding to reduce plant transient under voltage problems on the start-up transformer.

Safety Evaluation Summary

The modification resulted in increased protection against equipment failures and accidents by improving transient under voltage problems, resulting in an increased margin of safety.

SFC-81-119

This Specification Field Change covered the replacement of Palisades original low pressure turbine spindle.

Safety Evaluation Summary

The replacement spindle was specifically designed to lower the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety. The design alterations in the new rotor do not allow for a different type of accident or malfunction than any previously evaluated in the FSAR.

SFC-81-148; SFC-81-149; SFC-81-150; SFC-81-151; SFC-81-152; SFC-81-153; SFC-81-154; SFC-81-155; SFC-81-156; SFC-81-157; SFC-81-158 (EEQ)

The modifications done under the above SFC's upgraded electrical equipment to meet the present NRC environmental qualification requirements and guidelines per SEP Topic III-12.

Safety Evaluation Summary

The modifications made to the safety related electrical systems bring the systems up to the environmental qualification requirements established by the NRC (DOR guidelines and NUREG 0588), and in Technical Specification 6.14. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR has not been increased.

SFC-81-202

This change covered the provision of seismic braces for diesel generator fuel oil day tank to reduce to acceptable levels the stresses on both the tank walls and the tank supports during seismic conditions.

Safety Evaluation Summary

The accident or malfunction considered is failure of tank during seismic events. Due to this modification the probability of a tank wall rupture or shear of a tank support is reduced.

SFC82-113

This SFC covered the replacement of a block wall with reinforced concrete.

Safety Evaluation Summary

The replacement of an exterior block wall with reinforced concrete reduces the probability of an accident due to tornado related missiles. The wall will act as an integral portion of the existing auxiliary building wall.

SFC-82-186

This modification extended the inner personnel air lock door test connection to the exterior of the outer door bulkhead.

Safety Evaluation Summary

Extending the inner door test connection to the exterior of the outer door bulkhead does not increase the probability of an accident occurring. The consequences of an accident are not increased since the design for extending the test connection maintains a double isolation barrier between the containment interior and exterior atmospheres, thus radiological leakage from containment during an accident would not be increased.

SC-83-013

This change involved the replacement of threaded joints in 3" air lock pressurizing line with welded joints between valve P5A and the outer air lock wall.

Safety Evaluation Summary

The consequences of an accident due to radiological leakage are actually decreased because the replacement isolation barrier for the air supply line has no threads on the inboard side of the barrier. The replacement blind flange arrangement incorporates O-rings and a test port/plug so that testing can be performed after reinstallation of the flange to verify adequacy of the leak barrier. The design of the replacement barrier and replacement parts are of equal or better quality than original parts. The replacement does not reduce the margin of safety as defined in any Technical Specification basis, and does not create the possibility for a different accident type than any evaluated previously in the SAR.

SC-83-041

This covered the modification of penetration cooling ducts and temperature element locations on main steam and feedwater penetrations, changing the supply to V-50A and B.

Safety Evaluation Summary

The FSAR contains an analysis with the penetrations having no cooling. The additional cooling provided by sending air into the penetrations will lessen the possibility of concrete failure at the liner. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

SC-83-050

This modification involved a change in configuration of interlock locking pawl and orientation of gear assembly for personnel air lock.

Safety Evaluation Summary

The orientation of parts changed under this Specification Change does not affect operability or sealing capabilities of the air lock door. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

SC-83-107

This Specification Change covered the replacement of a 2" gate valve which showed evidence of leakage around the bonnet.

SC-83-107 - Continued

Safety Evaluation Summary

The replacement valve was of a better quality than the one being replaced. The overall safety and reliability of the valve is increased, and the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased, rather decreased.

SC-83-144

This modification covered the removal of a cap from the LPSI pump casing drain which was welded. The cap was cut off to drain or flush the pump, and a gate valve was added. This is a radiation hot spot.

Safety Evaluation Summary

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

SC-83-159

This change covered the revision of Specification 5935-C-53 to make ASME Code requirements for containment access openings consistent with current plant criteria as follows: The design, construction, inspection, and testing of the personnel and equipment openings and their attachments to reinforcement around such openings shall apply with Subsection NC of Section III of Nuclear Vessels, of the ASME Code as a minimum requirement instead of Subsection "B".

Safety Evaluation Summary

Reg Guide 1.26 does not require containment access openings to be put under Quality Group A classification (Class I components). Actual ASME Code requirement for air lock vessels is Subsection NE of ASME Section III. The change specified for Specification C-53 is consistent with code applicability as defined by current plant criteria also. (Quality Group B, ASME Section III for Class I components.) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased by this specification change.

SC-83-162

This Specification Change covered the modification of an isolation barrier for personnel air lock air supply line to allow for more convenient means of pressurizing the personnel air lock for a Tech Spec test.

SC-83-162 - Continued

Safety Evaluation Summary

The consequences of an accident due to radiological leakage are not increased due to this modification. The gate valve which was added is an improvement over the original design and allows proper pressurization of the air lock during the Tech Spec test to verify acceptable LLRT leakage.

SC-83-173

This change covered the modification of the equipment hatch test connection line.

Safety Evaluation Summary

The stronger test tubing material and relocated tubing run provide for greater resistance to tubing failure. The addition of test connection valve allows the plant to be more in conformance with the new ANSI-ANS 56.8-1981 and is consistent with other plant penetrations. Improving this test connection does not reduce the margin of safety as defined in any Tech Spec basis, and the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

SC-83-186

This modification replaced the pressurizer pressure transmitters.

Safety Evaluation Summary

The new transmitters are functionally equivalent to the prior transmitters which were replaced. The set points are still conservative. Installation was done during a refueling shutdown to prevent the possibility of a LCO while performing work. Equipment reliability has been increased by using new instruments with proven reliability. The old equipment had unpredictable drifting, thus the new instruments will add to the existing margins of safety and reliability.

SC-84-007

This Specification Change covered the shaft replacement on the main steam isolation valve.

Safety Evaluation Summary

The new shaft design and material increases the ductility and reduces the stress concentrations induced by sharp/abrupt geometrical changes in the keyway area, which should mitigate future cracking problems since the shaft is designed to absorb energy level experienced during a faulted condition. (A guillotine break on the valve discharge while at full steam flow.) The shafts

SC-84-007 - Continued

are functionally equivalent substitutes. Because of the above, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased, and the margin of safety is not reduced.

SC-84-125

This Specification Change covered the addition of a ½" spacer in the MSIV shaft. The additional spacer was installed because of the difference of measurement of the shoulder from the actuator retention nut.

Safety Evaluation Summary

The spacer was added to insure that the actuator would still be located on the shaft in the same position as it was previously. It was fabricated and installed upon vendor recommendations, review and approval of MSIV shaft alteration mods to mitigate linear indications and/or reduce stress concentrations on the shaft. The new assembly should improve the shaft, thus the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

Special Test T-95 - Initial Approach to Critical for a New Palisades Core

This Special Test provides surveillance for the initial approach to criticality after a core alteration.

Safety Evaluation Summary

The operational steps performed within this test are bounded by present safety analysis for the original and subsequent new cores. The procedure is used in conjunction with approved operating procedures and the result is additional surveillance and control of the critical approach. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

Special Test T-141 - Low Power Test Program for a New Palisades Core

This approved plant procedure was followed to obtain or verify certain reactor parameters and to compare these values with the Technical Specification requirements and predicted values.

Safety Evaluation Summary

The procedure prescribes activities which are within the limits and scope of 3.10.7 of the plant Technical Specifications entitled "Low Power Physics Testing." The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

Special Test T-142 - Base Power Level Selection

This procedure was used to determine an acceptable power level range for the performance of low power physics tests.

Safety Evaluation Summary

This procedure was an approved plant procedure, prescribing activities which are within the limits and scope of 3.10.7 of the plant Technical Specifications entitled "Low Power Physics Testing." The possibility of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased, and the margin of safety is not reduced.

Special Test T-143 - Zero Power Isothermal Temperature Coefficient Measurement

This procedure was used to obtain a measured value of isothermal temperature coefficient at near zero power and to compare it with predictions.

Safety Evaluation Summary

This low power test was performed under an approved plant procedure which describes activities which are within the limits and scope of 3.10.7 of the plant Technical Specifications entitled "Low Power Physics Testing." The possibility of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased, and the margin of safety is not reduced.

Special Test T-144 - Zero Power Rod Worth Measurements

This approved plant procedure was followed to measure the worth of various control rods in different configurations and to compare with predicted values.

Safety Evaluation Summary

This low power test was performed using a procedure prescribing activities which are within the limits and scope of 3.10.7 of the plant Technical Specifications entitled "Low Power Physics Testing." The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased. The margin of safety is not reduced.

Special Test T-145 - Zero Power Symmetry Check

This procedure was used to determine the degree of symmetry of the core.

Special Test T-145 - Zero Power Symmetry Check - Continued

Safety Evaluation Summary

This low power test was performed using an approved plant procedure which prescribes activities within the limits and scope of 3.10.7 of the plant Technical Specifications entitled "Low Power Physics Testing." The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased, and the margin of safety is not reduced.

Special Test T-163 - Fuel Sipping in Spent Fuel Pool

This approved plant procedure describes the steps necessary to perform fuel sipping.

Safety Evaluation Summary

The fuel pool boron concentration was maintained by controlling the demineralized water supply to the sipping system with a resettable metering valve. Sip system drains were routed directly into the fuel pool skimmers. The sip system automatically advances to a gas purge if the detector count rate exceeds 400 cps. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased and the margin of safety as defined in the basis for Technical Specification 4.2.1-6 was not reduced.