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DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT -
REACTOR PROTECTION SYSTEM MODIFICATION AND TECHNICAL
SPECIFICATIONS CHANGES (TAC NO. 66901)

Consumers Power Company letter of June 17, 1988 provided a partial response to the NRC request for additional information (RAI) concerning the Palisades Reactor Protection System modification and Technical Specifications Change Request (TSCR). Responses to the remaining questions are attached. Additionally, a copy of GAMMA-METRICS proprietary documents numbers 068 and 073 are attached. These documents were inadvertently omitted from our March 25, 1988 submittal. Permission for the NRC to copy these proprietary documents for internal distribution was given in Consumers Power Company letter of April 7, 1988.

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Attachments

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ATTACHMENT

Consumers Power Company
Palisades Plant
Docket 50-255

RESPONSES FOR REACTOR PROTECTION SYSTEM MODIFICATIONS RAI

(Items 2, 3, 4, 6 thru 11, and 22)

June 27, 1988

9 Pages

MI0688-0054-NL04

RESPONSES FOR REACTOR PROTECTION SYSTEM MODIFICATIONS RAI

2. Discuss any differences between the proposed Palisades modifications for the variable high power trip (VHPT) axial shape index (ASI) alarm, and thermal margin/low pressure (TM/LP) trip and those described in CENPD-199.

   Response: The VHPTtrip, TM/LPtrip and ASI alarm functions are the same as in CENPD-199(NP) except for the ASI alarm. The ASI function in the CE design is a trip function. Also the ASI from each excore detector string provides a signal to a single RPS channel, whereas, the CE design provides an averaged ASI signal to each RPS channel.

3. Discuss the acceptability of the nodal modeling in the modified version of PTSPWR2 for analyzing the asymmetric flow situations in CE split cold let (2x4) plants such as Palisades.

   Response: PTSPWR2, Advanced Nuclear Fuels Company (ANF) plant transient simulation computer code, has been modified to accommodate Combustion Engineering (CE) type 2x4 plant configurations. The CE 2x4 plant geometry consists of two primary coolant system (PCS) loops each with one hot leg and two cold legs. Each PCS loop contains one steam generator.

   The PTSPWR2 code has been updated to explicitly model each hot leg and each cold leg. Pipe characteristics (volume, loss coefficient, inertance, etc.) and PCS pump conditions can be individually specified for each leg. That is, for each of four cold legs a pipe volume, an initial PCS pump operating condition, a loss coefficient and an inertance is specified.

   Similar code updates were employed in the safety analysis performed for St. Lucie Unit 1. These updates are described in the Staff approved documentation for PTSPWR2 (Reference 1).

   The nodal modeling in the modified PTSPWR2 version is acceptable for analyzing asymmetric flow situations in CE 2x4 plants for two reasons. First, in general, PTSPWR2 is used to analyze the system thermal-hydraulic response for a variety of anticipated operational occurrences (AOO) and postulated accidents (PA) in which saturated two-phase PCS flow is precluded. The code evaluates the PCS flow, temperature, pressure and core power as a function of time. During these events, asymmetries in thermal-hydraulic conditions can exist due to the assumed conditions prior to the transient or as a result of the progression of the transient. Since the PCS remains in a subcooled state, the flow can reasonably be assumed to be incompressible so that detailed nodalization of the PCS is not necessary to evaluate the state equations of mass, momentum and energy. PTSPWR2 is not used for analysis of large or small break LOCA events or steam line break transients.

   Secondly, the results of the benchmark calculations, discussed in the response to Comment No. 4, serve to verify the nodalization of the PTSPWR2 CE 2x4 version. The transient event that was simulated for these benchmark calculations is an asymmetric pump coastdown.

Therefore, the nodal modeling in the modified PTSPWR2 version is acceptable for analyzing asymmetric flow situations in CE 2x4 plants.

4. Discuss the benchmarking of the revised version of PTSPWR2 against RELAP5/MOD2 in more detail. Describe the transient analyses which were compared and the differences in results, if any.

Response: The revised version of PTSPWR2 includes coding changes to allow a plant with CE split cold leg (2x4) geometry to be modeled. The purpose of the coding changes is to allow the analysis of asymmetric flow transients that affect only one cold leg or cold legs in opposite loops. The changes primarily are associated with increasing single dimension arrays to two dimensions.

The revised code has been benchmarked to a RELAP5/MOD2 simulation of a transient event with an asymmetric coolant flow distribution. The benchmark transient was a three primary coolant pump (PCP) coastdown event from 50% of rated power for a CE 2x4 plant.

The initial steady-state condition assumed three PCPs to be operating with a positive flow rate. The fourth PCP was modeled to be idle with reverse flow in the affected cold leg. The transient initiates with the coastdown of the three active PCPs. A primary coolant low flow trip occurs at 1.5 seconds. The pump speed of the three affected pumps is about 70% of rated after 5 seconds. During this time, as the positive suction head is decreasing in the coasting down pumps, the flow rate through the inactive loop is becoming more positive. The total simulation time was 50 seconds.

The results from the PTSPWR2 simulation closely match those of RELAP5/MOD2. The mass flow rates through each of the hot and cold legs are virtually identical throughout the event. The core exit temperature from RELAP5/MOD2 is slightly higher (approx. 1 to 5°F) than PTSPWR2. The maximum difference in the vessel outlet temperature is less than 4°F. The cold leg temperatures for the steam generator with the active PCPs are negligibly different while the PTSPWR2 predicted cold leg temperature for the steam generator with the idle loop is about 4°F higher. The differences in core inlet temperature are small (less than 1.5°F). The pressurizer pressure for RELAP5/MOD2 and PTSPWR2 are essentially the same to about 10 seconds, after which, the PTSPWR2 pressure decreases at a slightly greater rate. The difference in pressurizer pressure is, however, less than 55 psia at 50 seconds. The revised PTSPWR2 code results agree closely with those calculated with RELAP5/MOD2 for a three PCP coastdown event with one initially idle PCP.

In general, the benchmark calculations of PTSPWR2 (2x4) to RELAP5/MOD2 for a CE 2x4 asymmetric pump coastdown event provide a comparison as good as the PTSPWR2/RELAP5 benchmark calculations given in the Staff approved PTSPWR2 benchmark documentation (Reference 1) for a three pump coastdown event at H. B. Robinson Unit 2. Therefore, it is judged that the coding changes to allow the modeling of 2x4 CE plants in PTSPWR2 are adequate and applicable to Palisades.

6. Where are References 27, 28, 29 in ANF-87-150, Vol. 2?

   Response: In Section 15.0.7.2 (page 35), "References 29" should read "Reference 23". In Section 17.0.7.2.1 (page 36), "Reference 28" should read "Reference 22". Reference 27 was deleted from ANF-87-150(NP), Volume 2.

7. Describe how the specific TM/LP trip uncertainties described in CENDP-199, Rev. 1, are incorporated into the TM/LP uncertainties shown in Tables 15.0.7.2-1 of ANF-87-150, Vol. 2.

   Response: The uncertainties in the TM/LP trip function for Palisades have been accounted for in a deterministic manner. The specific uncertainties that have been applied include:

   - Instrument Drift (Power, T_{inlet})
   - Calorimetric Power
   - T_{inlet} Measurement
   - Pressure Measurement
   - RTD Measurement
   - Axial Shape Index

   A pressure bias of 165 psi was used to account for the pressure equivalent of instrument drift, calorimetric power, T_{inlet} measurement, pressure measurement and RTD measurement uncertainties. This equivalent pressure bias was conservatively used in the TM/LP trip development and the PTSPWR2 simulation of the transient events to maximize the challenge to the DNB SAFDL.

   The axial shape index (ASI) uncertainty of ±0.06 includes allowances for shape annealing, rod shadowing, signal processing and incore to excore calibration. The ASI uncertainty of ±0.06 was used to conservatively adjust an incore calculated ASI to an excore ASI. This excore ASI was then used in the TM/LP trip development, the T_{inlet} LCO development and the transient event analysis. These are the standard ASI uncertainties used for CE 2x4 plants.

   In addition, a transient allowance for T_{inlet} measurement delay was accounted for in the development and verification of the TM/LP trip. A t_{inlet} transient time delay of 1.5°F was used in the development of the TM/LP trip function. This value is based on the transient temperature response to a control rod withdrawal event and the time delay characteristic of the RTDs at Palisades.

   TM/LP trip processing and control rod drive holding coil delays were also included in the transient event analysis.

   The execution of specific transient events confirm the acceptability of the TM/LP trip function, if, when the uncertainties are counted for, the DNB SAFDL is shown not be to violated. ANF analyses include loss of load, control rod withdrawal and excess load transients to test the acceptability of the TM/LP trip.

8. Aside from the proposed reactor protection system modifications, describe any differences in methodology or assumptions between the reanalyzed Chapter 15 events presented in AFN-87-150 and the previous analyses of record.

Response: Aside from changes initiated by the installation of the modified reactor protection system, the following differences in major assumptions and methodology exist between the current submittal and the previous analysis of record. In general, the primary differences pertain to the use of new models in the transient analysis.

1). The PTSPWR2 version that is used incorporates several model changes, including a more realistic pressurizer model and homologous pump curves. These models have been approved by the NRC in Reference 1. An additional modeling change for the asymmetric flow transients is the capability of modeling 2x4 CE split leg geometry.

2). The NRC approved XCOBRA-3C computer code (Reference 2) includes the capability of using automated crossflow boundary condition information for each axial node in assembly subchannel calculations that are saved from a core calculation. This methodology has been reviewed and approved by the NRC.

The previous analysis of record incorporated a constant average flow rate in the subchannel calculation based on the crossflow from a core calculation.

3). Instead of the W-3 correlation, ANF's XNB critical heat flux correlation (Reference 3) is used to assess minimum DNBRs for each DNB limited event. This CHF correlation is NRC approved and has been shown to be applicable to Palisades in Reference 4.

4). The new analysis has assumed increased steam generator tube plugging to 29.3%, relative to 25.8% for the current analysis of record. The plugging level for Palisades Cycle 7 was approximately 24.6%.

5). In accordance with ANF's transient analysis methodology, minimum DNBRs for the current submittal are conservatively calculated using PTSPWR2 core conditions at the time of minimum DNBR in a steady-state XCOBRA-3C model.

The minimum DNBRs for the previous analysis of record were evaluated using PTSPWR2 core conditions as a function of time in a transient XCOBRA-3C model.

6). For the current submittal, the NRC approved SLOTRAX computer code (Reference 5) is used to assess the long term heat-up effects of a loss of normal feedwater event.

7). For the current analysis, the change in radial peaking factor limit as a function of power level is evaluated using the following relationship:

$$F_r(P) = F_r(100\%)[1. + .3(1 - P)]$$

where P is the fraction of rated power and $F_r(100\%)$ is the Technical Specification full power radial peaking limit.

The Palisades Technical Specifications are being revised to reflect the above radial peaking limit function.

The methods and major assumptions used in the current submittal are consistent with ANF's present transient analysis methodology and are deemed to be appropriate for the Palisades Plant (Reference 6).

9. Your proposed disposition of the CEA withdrawal event from 10^{-4} percent of rated power with 3 pumps operational allows limited fuel failure. Justify this with respect to General Design Criterion (GDC) 25 which requires the protection system to be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control system such as accidental withdrawal of control rods. GDC 25 makes no references to any probability assumption allowances. Also, since operation at hot shutdown with 3 pumps is allowed, discuss the consequences of a CEA bank withdrawal from hot shutdown or below.

Response: Proposed Palisades Technical Specification 3.1.1(b) (Reference 7) states that: "Four primary coolant pumps shall be in operation whenever the reactor is operated continually above hot shutdown". TS 3.1.1(b) also indicates that operation of three primary coolant pumps is for a limited amount of time. The allowance of three pump operation is to provide a limited time for repair/pump restart, to provide for an orderly shutdown or to provide for the conduct of reactor internals noise monitoring test measurements.

Operation with three primary coolant pumps is not considered to be a normal operating mode since it is allowed only for a short period of time and continual operation in this state is prohibited. A control rod bank withdrawal transient from a three pump initial condition is, therefore, considered to have a low probability of occurrence. As such, this event with three operating primary coolant pumps was classified as a Condition III or a Condition IV event.

The analysis for Event 15.4.1, as presented in ANF-87-150(NP), Vol. 2, conservatively bounds the consequences of a control rod bank withdrawal transient from hot shutdown and above with three primary coolant pumps operating. The results bound the event with four coolant pumps operating. For a cold shutdown initial operating mode, the maximum primary coolant temperature is $\leq$ 210°F, as compared to > 525°F for hot shutdown. From a DNB standpoint, initiating the event from hot shutdown is most limiting due to a higher primary coolant temperature. For refueling and hot shutdown operating modes, the required shutdown margin is sufficient to keep the core from becoming critical as a result of a control rod bank withdrawal.

10. What audible alarms are available during startup to alert the operator in the event of an inadvertent boron dilution?

Response: The following audible alarms are available during startup:

1. Pressurizer high level, after a bubble is established.

2. Source range startup rate alarm bistable light (not audible).

3. Log-N alarm and trip (10^{-4} to 15% power).

4. Volume Control Tank high level alarm and visual indications that CV-2155/CV-2165 is/are open (primary makeup tank source of unborated water).

5. T-ave/T-ref deviation alarm if MSIVs are closed.

11. How have the effects of 29% steam generator tube plugging been accounted for in the (1) increase in steam flow, (2) loss of load, (3) CVCS malfunction resulting in boron concentration decrease, (4) inadvertent opening of steam generator relief or safety valve, (5) inadvertent opening of pressurizer pressure relief valve, (6) steam generator tube failure, and (7) large break LOCA events?

Response: The primary effects of increasing steam generator tube plugging to 29.3% are as follows:

1) The increased steam generator tube plugging results in an increase in flow resistance in the steam generator and, consequently, the primary coolant system. The increased resistance through the steam generators causes a decrease in the overall PCS flow rate, decreases natural circulation flow rates and impacts LOCA blowdown flow characteristics.

2) Increasing the number of plugged steam generator tubes results in a decrease of the tube region heat transfer area in the affected steam generators. To extract the same amount of primary-side heat, this reduction of heat transfer area necessitates an increase in the primary-to-secondary side temperature difference. A reduced secondary-side temperature leads to a corresponding decrease in secondary-side pressure.

3) As the number of plugged steam generator tubes increases, the primary-side steam generator tube volume proportionally decreases. The reduced tube volume leads to a decrease in PCS mass for a given temperature distribution, pressure and pressurizer liquid level. A reduced primary-side mass increases the magnitude of the primary coolant temperature derivative for a given rate of heat transfer.

Each of these effects of steam generator tube plugging have been incorporated into the current transient analysis supporting Palisades operation.

22. Is the integrity test (software requirements 3.2.2.2) the same as the self-test (3.1.7.1)? Describe the portions of the TMM that these tests do not check on startup or repower.

Under certain conditions the integrity test and the self-test are the same. This is based on the results of the validity check of the interrupt status and the data stored in battery backed-up RAM. This testing is described below.

The Thermal Margin Monitor has several types of diagnostic tests and self-tests which are performed.

As described in the Software Requirements Specification, Document No. 055, (previously submitted on 3/25/88), section 3.1.7.1, the Thermal Margin Monitor performs the following tests.

On initialization and start-up (application of power), the Thermal Margin Monitor performs the tests outlined in the Software Design Description, including 1) ROM test, which is a cyclic redundancy check of each byte except for the last two bytes of ROM and comparing the results with the contents of the last two bytes of ROM, 2) System RAM test, which is a test of the ability to properly read and write to each byte of RAM, and 3) Application RAM test, which is a test of the ability to properly read and write to each byte of the battery-backed RAM. Following this testing, all interrupt vectors are initialized, the applications parameters, including the pump table parameters are loaded into the battery backed RAM, the Floating point error handler is initialized, the periodic interrupts are set, and the Real Time Clock is initialized.

During operation, the Thermal Margin Monitor performs the above tests as a lowest priority background task, continuously testing all ROM and RAM. In addition to this testing, the highest priority task is in the routine called Errorhang, which continuously checks for any error interrupts from any of the software procedures. These checks include testing for errors from the mathematical computation, from the switch inputs, from the analog to digital converter, and from the continuous RAM and ROM testing. Upon any detection of errors, the Errorhang routine suspends all tasks, sets all trip outputs to their tripped condition, sets the VHPT and Phi Minus B analog outputs to zero and sets the PTrip analog output to maximum, clears the screen and prints ERROR and then the number of the computational error, and then halts.

As described in the Software Requirments Specification, section 3.2.2.2, upon repower, the Thermal Margin Monitor performs a cyclic redundancy check of the contents of the battery backed RAM and checks the interrupt status which is stored in the battery backed RAM. If the cyclic redundancy check is valid and the interrupt status is valid, the calculations are then resumed at the point at which they were suspended by the power failure. The testing is then limited to the operational testing described above. If the interrupt status is not valid, then the Thermal Margin Monitor initiates the cold start routine including all ROM and RAM testing, as described in Software Requirements Specification, section 3.1.7.1.

Other than described above, the remaining portions of the TMM are not tested on start-up or on repower. An example of a portion of the TMM that is not tested on start-up or on repower is the screen and CRT output information. The screen driver is verified as ready to accept data, as a part of the integrity test or as a part of the background diagnostics tests, depending on the results of the testing of battery backed-up memory, but the ability of the driver and the CRT to display that data is not checked.

References:

1) "Description of Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR): Code Benchmarks", XN-74-5(A), Supplements 1-6, Exxon Nuclear Company, October 1986.

2) "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation", XN-NF-75-21(A), Revision 2, Exxon Nuclear Company, January 1986.

3) "Exxon Nuclear DNB Correlation for PWR Fuel Designs", XN-NF-621(A), Revision 1, Exxon Nuclear Company, April 1982.

4) "Justification of XNB Correlation for Palisades", XN-NF-709, Exxon Nuclear Company, May 1983.

5) "SLOTRAX-ML: A Computer Code for Analysis of Slow Transients in PWRs", XN-NF-85-24(A), Exxon Nuclear Company, September 1986.

6) "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors; Analysis of Chapter 15 Events", ANF-84-73(P), Revision 3, Advanced Nuclear Fuels Company, May 1988.

7) Docket 50-255, License DPR-20, Palisades Plant, "Technical Specifications Change Request-- Reactor Protection System", March 25, 1988.