TAIPOWER'S TRANSIENT ANALYSIS METHODOLOGY FOR PRESSURIZED WATER REACTORS

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ABSTRACT

The methodology presented in this paper is a part of the "Taipower's Reload Design and Transient Analysis Methodologies for Light Water Reactors" developed by the Taiwan Power Company (TPC) and the Institute of Nuclear Energy Research. This methodology utilizes four computer codes developed or sponsored by Electric Power Research Institute: system transient analysis code RETRAN-02, core thermal-hydraulic analysis code COBRAIIIC, three-dimensional spatial kinetics code ARROTTA, and fuel rod evaluation code FREY. Each of the computer codes was extensively validated. Analysis methods and modeling techniques were conservatively established for each application using a systematic evaluation with the assistance of sensitivity studies. The qualification results and analysis methods were documented in detail in TPC topical reports. The topical reports for COBRAIIIC, ARROTTA, and FREY have been reviewed and approved by the Atomic Energy Council (AEC). TPC's in-house transient analysis methodology have been successfully applied to provide valuable support for many operational issues and plant improvements for TPC's Maanshan Units 1 and 2. Major applications include the removal of the resistance temperature detector bypass system, the relaxation of the hot-full-power moderator temperature coefficient design criteria imposed by the ROCAEC due to a concern on Anticipated Transients Without Scram, the reduction of boron injection tank concentration and the elimination of the heat tracing, and the reduction of reactor coolant system flow.

INTRODUCTION

The methodology presented in this article is a part of the "Taipower's Reload Design and Transient Analysis Methodologies for Light Water Reactors" developed by the Taiwan Power Company (TPC or Taipower) and the Institute of Nuclear Energy Research (INER) (Huang, Peng, et al, 1994, Huang & Yang, 1996). TPC intends to use this methodology to perform Final Safety Analysis Report (FSAR) licensing basis analysis for the Maanshan Units 1 and 2 and similar pressurized water reactors (PWRs). Maanshan units are Westinghouse 3-loop PWRs, rated at 2775 MWt. The core contains 157 fuel assemblies with a 17x17 rod array. The fuel design currently used is the Westinghouse optimized fuel assembly (OFA).

The flow scheme of the Taipower's PWR transient analysis methodology is illustrated in Figure 1. This methodology utilizes four computer codes developed or sponsored by Electric Power Research Institute (EPRI): system transient analysis code RETRAN-02 (McFadden, et al, 1988), core thermal-hydraulic analysis code COBRAIIIC (Jackson & Todreas, 1981), three-dimensional (3-D) spatial kinetics code ARROTTA (Eisenhart, 1991), and fuel rod evaluation code FREY (Rashid, et al, 1989). RETRAN-02 determines the transient system responses and provide transient core thermal-hydraulic state parameters to COBRAIIIC and FREY for departure from nucleate boiling ratio (DNBR) and peak clad temperature (PCT) calculations, respectively. The core physics and power profile related inputs to RETRAN-02 are generated by the Core Management System (CMS) developed by Studsvik of America, while the fuel rod performance related inputs are prepared by ESCORE. COBRAIIIC utilizes the core state parameters generated by RETRAN-02, and limiting power profile and gap conductance data to evaluate the fuel DNBR margin. ARROTTA performs 3-D core kinetics calculations for multi-dimensional core transients which can not be analyzed by RETRAN-02, such as the rod cluster control assembly (RCCA or rod) ejection accident. FREY utilizes the core power and core flow transients generated by RETRAN-02 or ARROTTA to evaluate the maximum fuel stored energy and PCT during the transient.

Two methodologies are used in the TPC's transient analysis. The Statistical Thermal Design Method (STDM) is used to analyze the Condition II departure from nucleate boiling (DNB) events initiated at power operation condition and the Condition III complete loss of flow event. The Conservative Bounding Analysis Method is used for non-STDM event analysis. For non-STDM event analysis, the input parameters are assumed at their worst values while the DNBR design limit equals to the correlation limit. For STDM event analysis, all the key input parameters are assumed at their nominal values while DNBR design limit is based on the statistical combination of the uncertainties of the key system parameters (including plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters) and the DNBR correlation.

Each of the computer codes was extensively validated by comparing with plant measured data, relevant test data, and vendor calculated results. Analysis methods and modeling techniques were conservatively established for each application using a systematic evaluation with the assistance of sensitivity studies. The qualification results and analysis methods were then documented in detail in topical reports and submitted to the Atomic Energy Council (AEC) for review and approval. The following sections describe the analysis methods and method qualification efforts, topical report licensing, and the major applications accomplished by the TPC.

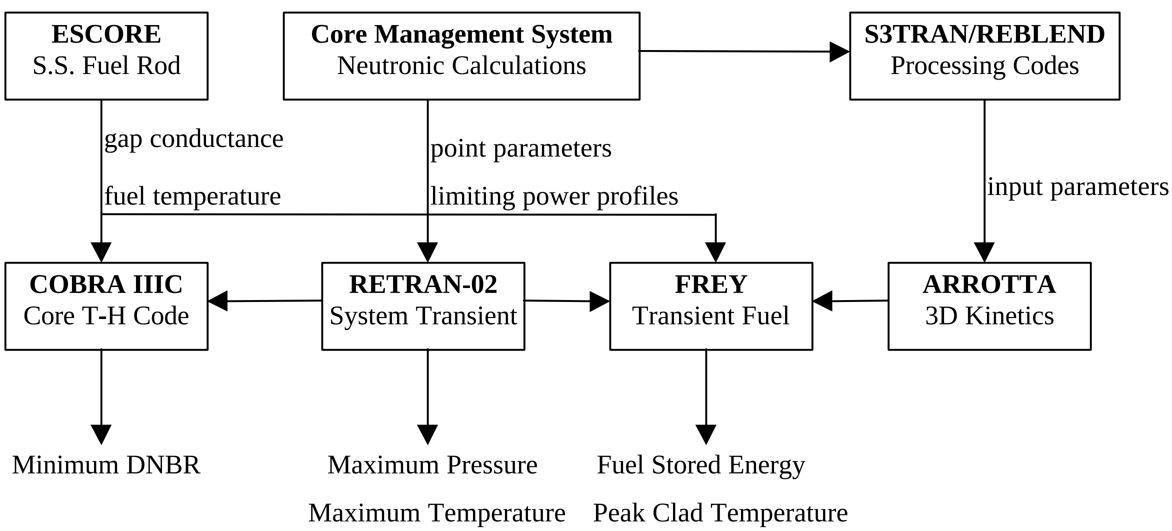

Figure 1 Flow Scheme for Taipower's PWR Transient Analysis Methodology

RETRAN SYSTEM TRANSIENT ANALYSIS METHOD

Method Description

RETRAN-02 is a large and sophisticated computer code developed by EPRI to simulate a wide spectrum of thermal-hydraulic transients for both PWRs and boiling water reactors (BWRs). A best estimate RETRAN model is developed to simulate the expected transient response, based on plant specific drawings and technical documents. The key modeling techniques and assumptions are:

1. The nodalization scheme contains sufficient numbers of volumes, heat conductors, and flow junctions to make adequate representation of the plant. Three loops are modeled explicitly to simulate imbalance conditions expected in certain asymmetric transients. In the usual case, the core is represented by three volumes with a single flow path, and the reactor coolant system (RCS) flows from the three loops are completely mixed inside the reactor pressure vessel. Specially for events with asymmetric core conditions such as main steam line break (MSLB), a split core model capable of modeling incomplete mixing at the downcomer and lower plenum is developed. The point kinetics model in RETRAN is selected to calculate the transient nuclear power response.
2. RETRAN control systems were developed and used extensively to specify transient boundary conditions, such as automatic plant actions and operator actions as well as control actions by modulating valves, changing fill rates or reactivity and simulation of trip actuation.
3. Code models and code options are properly selected to simulate various components; e.g., the non-equilibrium pressurizer option is used to model the pressurizer.
4. Reactor protection system (RPS) trip functions are simulated via RETRAN trip cards.
5. Plant control systems in general are modeled with RETRAN control blocks. The control systems modeled include automatic rod control system, feedwater flow control system, pressurizer pressure control system, and steam dump control system.
6. Two of the engineering safeguard features (ESFs), high head safety injection system and auxiliary feedwater system, are modeled.

The FSAR accident analyses in general are performed using a bounding analysis approach. To analyze the FSAR accident analyses, it is required to modify the best estimate RETRAN model described above to provide the needed conservatisms as well as to provide capability to simulate specific events. The major modifications include:

1. For most accidents which are DNB limited, nominal values of initial conditions are assumed. The uncertainties on power, temperature, and pressure determined on a statistical basis are included in the DNBR design limit. For accidents which are not DNB limited, or in which the STDM is not applicable, the initial conditions are obtained by adding the uncertainties to the nominal values.
2. The pertinent plant systems and equipment available for mitigation of each accident are modeled. For control systems, however, the action is modeled only if that action results in more severe consequences, i.e., no credit is taken for control system operation if that operation mitigates the event. For some events, the analysis is performed both with and without control system operation to determine the worst case. Similarly, no credit is taken for components which are not safety grade such as pressurizer and steam generator power operated relief valves, unless the action results in more severe consequences.
3. Conservative reactor trip and ESF actuation setpoints which delay the time of signal actuation are used.

4. Conservative reload related parameters are assumed such that reanalysis is not expected for all subsequent cycles. The reload related parameters modeled by RETRAN are those parameters required by the point kinetics calculations: moderator temperature coefficient (MTC), Doppler power coefficient (DPC), and control rod worth parameters (reactivity insertion following reactor trip and shutdown margin). The conservative directions for the MTC and DPC are identified and bounding values are assumed for each of the FSAR accidents. The reactivity insertion following reactor trip is a combination of a minimum available tripped worth and a normalized reactivity insertion rate which delays the reactivity insertion.

Method Qualification

To demonstrate the technical competence in using RETRAN and qualify the Maanshan RETRAN models for thermal-hydraulic simulation, extensive efforts have been performed for three major tasks:

1. Detailed evaluations of the plant nodalizations, control system models, code models, and code options selected. This task was accomplished by a systematic evaluation with the assistance of sensitive studies.
2. Validation against plant startup test data and plant operational transient data from the Maanshan plant. The Maanshan RETRAN model has been validated against: (1) 100% power large load reduction test, (2) 100% power turbine trip test, (3) 100% power net load rejection test, and (4) 100% power loss of main feedwater event. The validation includes the direct comparisons between the RETRAN calculated and the measured data for the recorded plant parameters. Significant variations were identified and potential causes are evaluated. The validation results indicate that RETRAN predicted transient results in general agree well with the test data in important parameters such as nuclear power, RCS temperature, pressurizer pressure, opening positions of the steam dump valves, and the control rod positions, etc. It can be concluded that Maanshan RETRAN model is capable of simulating various transients with a good accuracy.
3. Validation against vendor's results. Benchmark analyses have been performed for the following FSAR limiting transients with respect to various design criteria: (1) complete loss of flow for DNB, (2) turbine trip and locked rotor for RCS overpressurization, (3) loss of normal feedwater and feedwater line break for degradation in secondary heat removal capability, (4) MSLB for asymmetric cooldown which leads to recriticality, and (5) anticipated transient without scram. RETRAN results were compared against the vendor's analysis results, and causes for significant deviations were identified and justified. The validation results indicate that RETRAN predicted transient results in general agree very well with the vendor's results, if similar initial conditions and analysis assumptions are used. It is therefore concluded that Maanshan RETRAN model is acceptable for analyzing FSAR licensing basis transients.

COBRAIIIC CORE THERMAL-HYDRAULIC ANALYSIS METHOD

Method Description

The core thermal-hydraulic code COBRAIIIC determines the core thermal margin for both steady state and transient conditions. The COBRAIIIC version used is a modified version of COBRAIIIC/MIT-2. The WRB-1 DNB correlation used by all the DNB events in the Maanshan FSAR except MSLB was incorporated into the TPC's COBRAIIIC version. The COBRAIIIC modeling methodology utilizes the single stage or one-pass approach. The core is modeled with a number of flow channels of various sizes and shapes. The general concept is to model the hot subchannels and their surroundings in detail, and lump those far away from the hot channels together into large channels. This approach minimizes the number of channels and also preserve the accuracy of the DNB calculations. A one-eighth core 22-channel model, which consists of 9 subchannels and 13 lumped channels, was developed based on the concept and is applicable for all the DNB events with an exception of the MSLB event. Additionally, a full core 16-channel COBRAIIIC model was developed for the MSLB DNB calculation. Analysis methods and modeling techniques were conservatively established for each input parameter using a systematic evaluation with the assistance of sensitivity studies. Sensitivity studies were performed to conservatively establish the modeling techniques for radial noding, axial noding, crossflow parameters, turbulent mixing parameters, axial friction factor, two-phase flow correlations, and grid form loss coefficient. The following conservative assumptions used by the current FSAR licensing basis analysis are included: (1) a conservatively flat radial power distribution within the hot assembly, such that flow mixing around the hot channels is minimized, (2) a conservative axial power profile, typically a 1.55 chopped cosine, (3) a high fraction (97.4%) of power generated within the fuel rods, (4) a conservatively low turbulent mixing coefficient value of 0.038, and (5) a 5% reduction of coolant flow to the hot assembly.

Method Qualification

To justify use of the WRB-1 correlation with the COBRAIIIC code and determine the correlation limit, the Westinghouse rod bundle data (19 test series involving 970 data points) with different rod diameter, heated length, axial power shape, grid type and grid spacing were performed using the COBRAIIIC code. In general, the COBRAIIIC predicted results agree very well with the test data and vendor predicted results. Based on the measured to predicted critical heat flux ratio data, the 95/95 DNB correlation limits and the DNBR design limits for the WRB-1 correlation and the TPC/INER developed EPRI-1P correlation have been assessed. For the WRB-1 correlation, the calculated 95/95 correlation limit is 1.167, which is very close to the vendor’s value of 1.163. With DNB correlation uncertainty combined statistically with the uncertainties of the key system parameters, the resulting DNBR design limits are 1.234 and 1.222 for typical cell and thimble cell, respectively. For the EPRI-P correlation, the 95/95 correlation limit is 1.085, and resulting DNBR design limits are 1.119 and 1.120 for typical cell and thimble cell, respectively. Finally, to validate the Maanshan COBRAIIIC model, DNBR calculations were performed for the four cases: (1) nominal steady-state operating conditions, (2) the complete loss of flow transient, (3) thermal limit DNBR conditions, and (4) the MSLB conditions. The qualification analyses performed indicate that the DNBR results predicted by the COBRAIIIC model match very well with the vendor’s results; this further assures the adequacy of the COBRAIIIC modeling methodology.

ARROTTA CORE KINETICS ANALYSIS METHOD

Method Description

The main application of the 3-D kinetics code ARROTTA is to perform 3-D core kinetics calculations for multi-dimensional core transients which can not be analyzed by RETRAN-02, such as the rod cluster control assembly (RCCA or rod) ejection accident. The neutronic data for the input of ARROTTA are prepared by a series of interface codes (Huang, Yang, et al, 1994). The reactor core is modeled radially with one node per fuel assembly and at least two nodes of reflector. Axially, there are 14 nodes, in which the top and bottom nodes are reflector. The control rod is set to eject in 0.1 second from the initial position to the fully withdrawn position with constant speed. The ejected rod adds a positive reactivity to the initially critical reactor such that the neutron flux and the fission power increase rapidly. The reactor core is tripped by the insertion of other control rods after the trip signal is initiated by either high or low setpoint of the power range neutron flux. To avoid unnecessary reanalysis, the rod ejection analysis is performed with parameters which are expected to be bounding for any expected future reloads. All the important physics parameters used in the analysis are adjusted to the limit values presented in the Reload Safety Analysis Checklist (RSAC). Moreover, the analysis should bound all the operating conditions in Modes 1, 2, and 3. The event is not analyzed in Modes 4 or 5 since the probability of occurrence is lower and the reactor coolant system pressure is reduced, while in Mode 6 the event is prevented because the core has been shut down, and the reactivity insertion by a single rod ejection cannot bring it back to critical. The event is analyzed for four plant states: both hot-zero-power (HZP) and hot-full-power (HFP) for the beginning-of-life (BOL) and the end-of-life (EOL) conditions. The results will bound middle-of-life and part power conditions.

Method Qualification

To qualify ARROTTA for the application to TPC's licensing basis events, ARROTTA has been validated extensively for both static core and kinetic core analyses (Huang, Yang, et al, 1994). The static calculations compared include critical boron concentration, core power distribution, and control rod worth. The results indicate that ARROTTA predictions match very well with plant measured data for Maanshan and SIMULATE-3 predictions. The kinetic benchmark problems validated include the Nuclear Energy Agency Committee on Reactor Physics (NEACRP) rod ejection problem, the 3-D Langenbuch-Maurer-Werner light water reactor (LWR) rod withdrawal/insertion problem, and the 3-D linear regression analysis (LRA) BWR transient benchmark problem.

Four cases representing the conditions of BOL HFP, BOL HZP, EOL HFP, and EOL HZP were performed with ARROTTA. Figure 2 compares the ARROTTA calculated core power transient response for the EOL-HFP case with the vendor's result reported in the current Maanshan FSAR. The ARROTTA and vendor predicted core power transient responses are similar, however, the ARROTTA predicted peak power level is lower. It is noted that in the vendor's analysis, the 3-D kinetic code is used as an axial one dimensional code and conservative assumptions are employed for the radial dimension, i.e., a conservative weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature and no weighting is applied to the moderator feedback. These assumptions are expected to lead to a higher peak power level.

FREY HOT ROD ANALYSIS METHOD

Method Description

FREY is a computer program for the thermo-mechanical analysis of a single LWR fuel rod considering its detailed behavior under steady state and transient conditions. The main application of the FREY code is to perform the transient fuel rod stored energy and temperature calculations for fuel heatup transients such as rod ejection and locked rotor. Because of the rapid power rise and/or flow reduction, DNB condition is assumed in the fuel heatup analysis for the rod ejection and locked rotor accidents. The Bishop-Sandberg-Tong (BST) correlation is used by the current Maanshan FSAR analysis to determine the heat transfer coefficient after DNB. The coding modification for the BST correlation was incorporated into the TPC's FREY version. For the hot rod analysis method, the general concept is to maximize the initial fuel stored energy and energy addition, and minimize the energy release. The important assumptions include:

1. A conservative fuel-clad gap heat transfer. By adjusting the cold gap size between fuel and clad, the initial fuel temperature at the hot spot for the HFP cases is set to the limiting (maximum) hot spot temperature from the RSAC. Following the initiation of the event, the cold gap size is set to the as-manufactured gap size for the maximum fuel stored energy calculation , while gap closure is assumed to occur for the PCT calculation.
2. A minimum heat transfer to coolant. For conservatism, DNB is assumed to occur at the initiation of the locked rotor event. For the rod ejection analysis, DNB is assumed to occur at 0.1 second after the initiation of the transient for the HFP cases, and at the initiation of the event for the HZP cases.

3. A maximum total peaking factor (Fq). For the locked rotor analysis, Fq is assumed to be 2.5 throughout the complete transient; this value is conservative since the limiting value specified in the Technical Specifications is only 2.2 and Fq value is expected to decrease after the event initiation. For the rod ejection analysis, Fq is assumed to be 2.5 at the event initiation, and is linearly ramped to the RSAC limiting value in 0.1 second; the hot rod locations before and after rod ejection are assumed coincident.
4. A minimum initial RCS flow rate, which corresponds to the thermal design flow rate, is used.

Method Qualification

The qualification effort performed by TPC includes the comparison of FREY predictions to experimental data from BOSS-FC02 and reactivity initiated accident tests, as well as to vendor's results for rod ejection and locked rotor accidents for the TPC' Maanshan plant. The comparisons of the FREY predicted results with the experimental data indicate that FREY calculated fuel and clad temperatures in general agree well with the experimental data, if the cold gap is properly selected. Using the core power and flow transients reported in the current Maanshan FSAR for rod ejection and locked rotor accidents, the FREY predicted fuel and clad temperatures, and fuel enthalpy responses match the vendor's results very well. Figure 3 compares the FREY predicted results and vendor's results for the pertinent parameters at the hot spot for the EOL-HFP rod ejection case. Excellent agreement is obtained for the hot spot fuel center temperature and fuel average temperature, while the FREY predicted clad outer temperature is slightly lower. The FREY predicted maximum fuel stored energy is 165 cal/gm, which is very close to the vendor's value, 168 cal/gm.

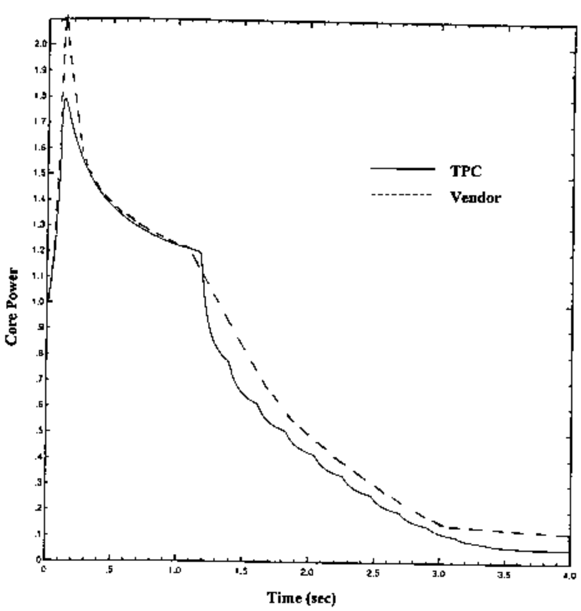

Figure 2 Comparison of Core Power Transient

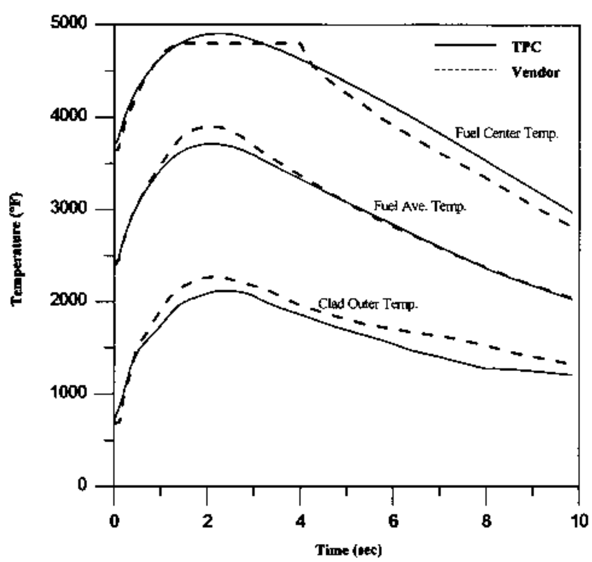

Figure 3 Comparison of Hot Spot Parameters

TOPICAL REPORT LICENSING

The qualification results and analysis methods have been documented thoroughly in four TPC topical reports (Huang & Chiu, 1996, Huang & Hsieh, 1995, Cheng et al, 1996, Kao & Huang, 1997). The topical reports for COBRAIIIC, ARROTTA, and FREY have been submitted to the ROCAEC, and formal approval has been obtained after all the review issues completely clarified by the TPC. Along with the two approved topical reports for the Taipower's PWR reload safety evaluation methodology (Huang & Wang, 1992, Huang, 1995), five topical reports for the Taipower's PWR reload and transient analysis methodologies have been approved.

CONCLUSIONS AND APPLICATIONS

The Taipower's PWR transient analysis methodology has been thoroughly examined. The qualification analyses performed indicate that TPC's analysis results in general agree very well with the measured data and vendor's results. Formal ROCAEC approval has been obtained for three topical reports. TPC's in-house transient analysis methodology has been successfully applied to provide valuable support for many operational issues and plant improvements for Maanshan Units 1 and 2. Major applications include:

1. Technical Specification changes for the removal of the resistance temperature detector (RTD) bypass system recently completed. Sensitivity studies were performed to assess the effects of a change in the spread of the RTD sensor response time and pure delay time from 4/4 seconds to 6/2 seconds, and the effects of a further increase in total RTD total response time from 8 to 10 seconds.

2. Relaxation of the HFP MTC design criterion imposed by the ROCAEC due to a concern on Anticipated Transients Without Scram (ATWS) following Chernobyl accident. The peak RCS pressure results calculated by RETRAN ATWS analyses justified a revision of the HFP MTC criterion at 1% exposure from symbol 163 \f "Symbol" \s 11≤} -7 pcm/symbol 176 \f "Symbol" \s 11°}F to symbol 163 \f "Symbol" \s 11≤} -4 pcm/symbol 176 \f "Symbol" \s 11°}F, which is needed for an extension of the fuel cycle length to 18 months.

3. Technical Specification changes for the BIT concentration reduction and the heat tracing elimination.

4. Technical Specification changes for a 2 % reduction in the RCS flow.

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KEY WORDS

Transient analysis, method, qualification, Taipower, pressurized water reactors, Taiwan Power Company.