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TITRAM FAST TRANSIENT LICENSING ANALYSIS METHODOLOGY FOR KUOSHENG BWR/6

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ABSTRACT

This paper presents the TITRAM (TPC/INER Transient Analysis Method) methodology for the fast transient analysis of Kuosheng Nuclear Power Station (KSNPS) with two units of General Electric (GE) designed BWR/6 (Boiling Water Reactor). The purpose of this work is to provide a technical basis of Taiwan Power Company(TPC)/Institute of Nuclear Energy Research (INER)'s qualification to perform plant specific licensing safety analyses for the Final Safety Analysis Report (FSAR) basis system fast transients, and related plant operational transient analyses for the Kuosheng plant. The major task of qualifying TITRAM as a licensing method for BWR transient analysis is to adequately quantify its analysis uncertainty. A similar approach as the CSAU (Code Scaling, Applicability, and Uncertainty Evaluation) methodology developed by the USNRC (United States Nuclear Regulatory Commission) was adopted. The CSAU methodology could be characterized as three significant processes, namely code applicability, transient scenario specification and uncertainty evaluation based on Phenomena Identification and Ranking. The applicability of the TITRAM code package primarily using the SIMULATE-3 and RETRAN-3D codes are demonstrated with analyses of integral plant tests such as Peach Bottom Turbine Trip Test and plant startup tests of KSNPS. A Phenomena Identification and Ranking Table (PIRT) with uncertainty values for each identified parameter to cover 95% of possible values are established for the selected KSNPS fast transients. The experience from BWR organizations in the nuclear industry is used as a guide in construction of the PIRT. Sensitivity studies and associated statistical analyses are performed to determine the overall uncertainty of fast transient analysis with TITRAM based on the KSNPS Analysis Nominal Model. Finally, the Licensing Model is established for future licensing applications.

INTRODUCTION

This paper addresses the TITRAM Methods for the fast transients initiated from the full power operational condition. It contains four transients:

For Thermal Limit, primarily Critical Power Ratio (CPR):
1) Feedwater Controller Failure Without Bypass (FWCFNB)
2) Turbine Trip Without Bypass (TTNB)
3) Load Rejection Without Bypass (LRNB)

For Pressure Limit:
4) ASME (American Society Of Mechanical Engineers) Overpressurization Transients including Turbine Stop Valve Closure (ASME-TSVC), Turbine Control Valve Closure (ASME-TCVC), and Main Steam Isolation Valve Closure (ASME-MSIVC)

The Topical Report[1] of this work was submitted to the licensing authority of Taiwan, AEC (Atomic Energy Council), for review in January, 2009, and received a Safety Evaluation Report in July, 2009. The method has been approved for licensing applications. There are three previous approved Topical Reports[2][3][4] documenting TITRAM, which support this work.

In order to apply TITRAM codes to licensing analyses, the uncertainty of the calculations has to be determined since the nominal results from the TITRAM codes are best estimate values. A similar approach to the CSAU[6] methodology is adopted in TITRAM. The 14 CSAU steps are characterized as three categories, namely code applicability, phenomena identification and ranking based on the transient scenarios, and uncertainty evaluation. Each of these three is applied in TITRAM for KSNPS and described in this paper.

THE CSAU METHODOLOGY

The CSAU methodology was developed by the USNRC, its contractors, and consultants in 1989 to address, in a unified and systematic manner, questions relating to:

1) the scaling capability of a best-estimate code,
2) its applicability to scenarios of interest to nuclear power plant safety studies, and
3) the evaluation of uncertainties in calculating parameters of interest when the code is used to perform a calculation for a specified scenario and plant design.

The NRC's report[6] applied the CSAU methodology to a Large-Break Loss-of-Coolant Accident. However, it outlined a rigorous process for how to apply best estimate codes and how to quantify the overall model and plant parameter uncertainty. The CSAU methodology is therefore applicable to other event scenarios such as Anticipated Operational Occurrences (AOO). The work of Tokyo Electric Power Company on TRACG[6] to licensing analysis is one of the examples applying CSAU in the transient analysis.

Table 1 Code Scaling, Applicability and Uncertainty Evaluation Methodology

Step	Description
1	Scenario Specification
2	Nuclear Power Plant Selection
3	Phenomena Identification and Ranking
4	Frozen Code Version Selection
5	Code Documentation
6	Determination of Code Applicability
7	Establishment of Assessment Matrix
8	Nuclear Power Plant Nodalization Definition
9	Definition of Code and Experimental Accuracy
10	Determination of Effect of Scale
11	Determination of the Effect of Reactor Input Parameters and State
12	Performance of Nuclear Power Plant Sensitivity Calculations
13	Determination of Combined Bias and Uncertainty
14	Determination of Total Uncertainty

Table 1 outlines the 14 steps of the CSAU methodology. These 14 steps could be characterized as processes showing the code applicability with definition of model uncertainties, nodalization and scale uncertainties and plant parameter uncertainties, quantification and combination of uncertainties based on a Phenomena Identification and Ranking Table (PIRT). The following key areas could be used to cover the 14 steps, namely:

1) Code Applicability,
2) Transient Identification and Phenomena Identification and Ranking,
3) Uncertainty Evaluation.

CODE APPLICABILITY

The capability of a code or a series of codes to calculate an event for a nuclear power plant depends on four elements:

1) Conservation equations, which provide the code capability to address global processes,
2) Constitutive correlations and models, which provide the code capability to model and scale particular processes,
3) Numerics, which provide the code capability to perform efficient and reliable calculations, and
4) Structure and nodalization, which address code capability to model plant geometry and perform efficient and accurate plant calculations.

These four elements must be considered when evaluating the applicability of the code to the transient of interest for the nuclear power plant calculation. The evaluation of the capability to calculate the selected transient events can be done by performing comparisons with separate effects tests, integral effects tests and full scale plant data.

The TITRAM Code Package

The TITRAM computer codes include MICBURN-3[7], CASMO-3[8], TABLES-3[9] SIMULATE-3[10], SLICK[11], BWRHB-INER[12], RETRAN-3D[13], and AutoDCPR[14], a program for Critical Power Ratio calculations. Figure 1 shows the analysis flow of TITRAM. The applications of these codes are under a quality assurance program[15] corresponding to the requirement of USNRC Regulations 10 CFR Appendix B to Part 50--Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

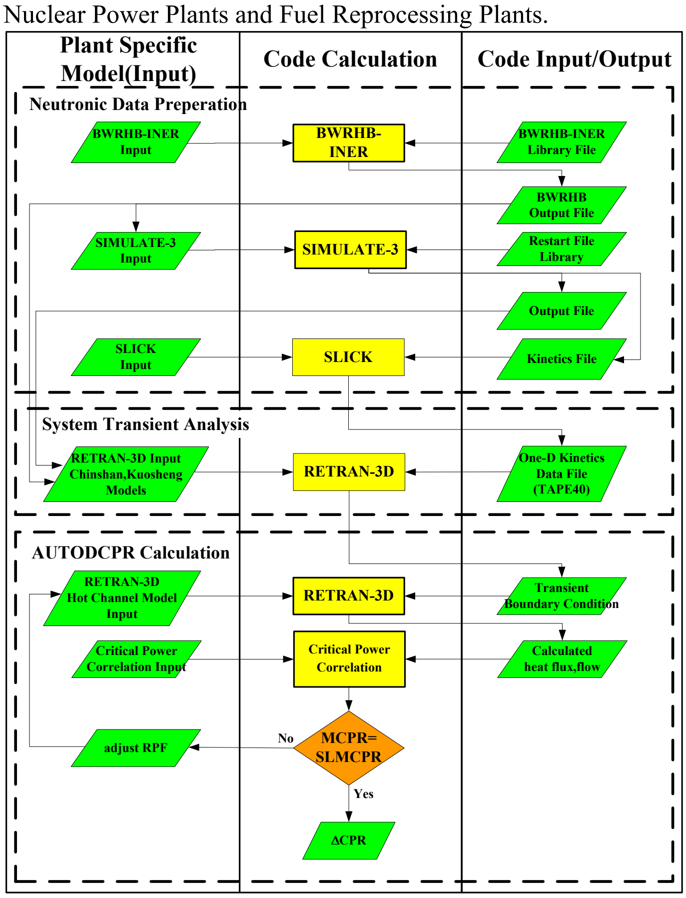

Figure 1 TITRAM Analysis Flow

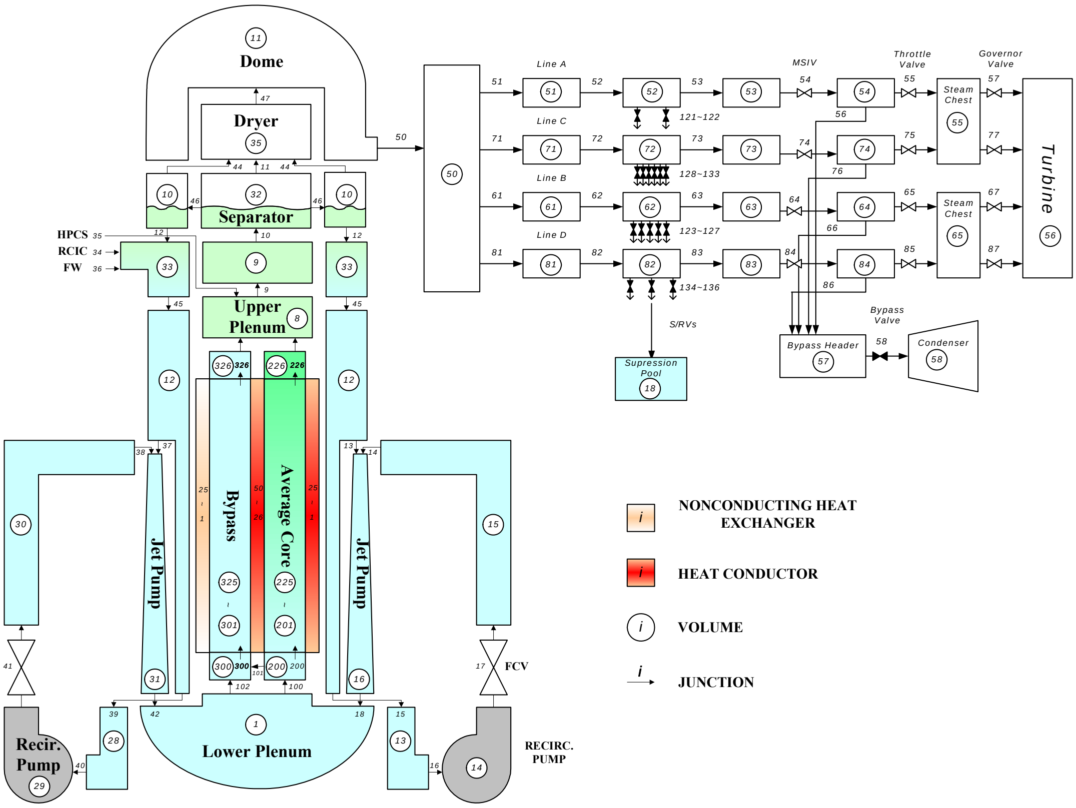

Figure 2 Kuosheng RETRAN-3D Model

The MICBURN-3, CASMO-3, TABLES-3, SIMULATE-3 and SLICK codes developed by STUDSVIK of America (SOA), Inc., are used to prepare power distribution and kinetics data for the system transient analysis with RETRAN-3D. SIMULATE-3 is an advanced three-dimensional two-group, coarse mesh diffusion theory core analysis code which employs higher-order spatial flux representation and an advanced fuel assembly model, with explicit reflector treatment. SLICK reads data from a SIMULATE-3 output file and generates cross-section library specifically for RETRAN-3D one-dimensional kinetics calculation. These codes and models to produce input to for transient analysis were reviewed for various applications[16].

Based on energy and mass conservation equations, BWRHB-INER[12] calculates thermal-hydraulic parameters such as steam dome pressure, core inlet subcooling, steam flow for the use in SIMULATE-3 and RETRAN-3D models.

The TITRAM methodology for system transient thermal hydraulics analysis is based on the Electric Power Research Institute (EPRI) RETRAN-3D[13] code. RETRAN-3D, developed by Computer Simulation & Analysis, Inc. (CSA), is the latest generation RETRAN code representing an advanced and robust code with extended analysis capability. It is an extension of the RETRAN-02[17] transient thermal-hydraulic analysis code designed for use in best-estimate evaluation of light water reactor systems.

RETRAN-3D was submitted for USNRC review by the RETRAN Maintenance Group on July 8, 1998 and received the SER[18] in January of 2001. The newly released (March 9, 2006) CSA's QA version of MOD. 4.2 of the RETRAN-3D code is used.

The AutoDCPR[14] is composed of a RETRAN-3D hot-channel analysis which drives the critical power correlation subroutine, in an iterative process to determine the minimum ΔCPR. AutoDCPR links the RETRAN-3D system model (Figure 2) and hot-channel model to calculate transient ΔCPR.

The Evaluation of TITRAM Capability

The applicability of the TITRAM code package primarily using SIMULATE-3 and RETRAN-3D is demonstrated with analyses of integral plant tests such as the Peach Bottom Turbine Trip Test, KSNPS plant startup tests and real plant transients.

The Peach Bottom Turbine Trip Test (PBTT) was analyzed using TITRAM. The purpose is to demonstrate the adequacy of the codes especially the linkage between SIMULATE-3 and RETRAN-3D and the modeling techniques and options also be used in the Kuosheng RETRAN-3D model. PBTT benchmarking has been widely used as a demonstration of code applicability because the turbine trip tests are full scale BWR tests which resulted in significant neutron flux peaks, similar to the selected fast transients. The direct scram on the Turbine Stop Valves (TSV) position 10% closed was disabled to produce the spikes. This test was so important to the industry that it was the basis for an international code benchmark effort sponsored by the USNRC and OECD[19] (Organization for Economic Co-Operation and Development).

The active fuel channel of RETRAN-3D PB plant model is axially divided into 24 axial nodes to be the same as the upstream PB SIMULATE-3 model. A best-estimated scram speed was modeled as specified by the OECD benchmark specification. Comparison of the measured averaged LPRM signals with the calculated results is shown in Figures 3. The LPRM result is normalized to the initial condition of 61.6% of rated flux. The RETRAN-3D results show good agreements with the measured data in terms of timing and trend but with a higher peak neutron flux, demonstrating the conservatism of the TITRAM RETRAN-3D model.

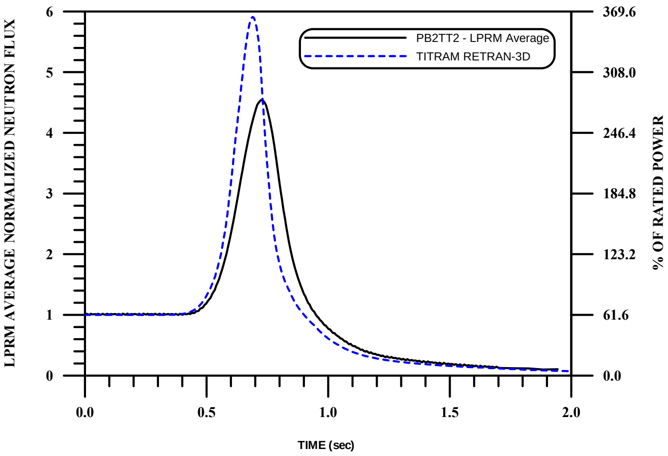

Figure 2 PBTT2 LPRM Results

Table 2 shows the comparison between PBTT test results and the calculated results including the integral power (IP). The IP of TITRAM RETRAN-3D results is larger than that of the test. Hence, based on an analysis of an integral effects test with full scale plant data, TITRAM method shows its applicability with enough conservatism in licensing pressurization transient analyses.

Table 2 PBTT2 Measured/Calculated Results

PBTT2	Measured	TITRAM RETRAN-3D
Time of Peak (sec)	0.726	0.690
Peak Power (normalized)	4.547	5.916
Ratio (Calculated/Measured)	1	1.3
Integral Power (MW-sec) Rise-high Portion	1518.57	1652.49
Integral Power (MW-sec) The Whole 2sec	4034.27	4236.68

In addition, a series of plant startup tests analyses[3] were conducted to help evaluate the code applicability. Two tests were done in 1981 for Kuosheng Unit 1, and the other two tests were performed in 1982 for Unit 2. The analyses showed good agreement between the calculated results and the plant test data. These are:

1) 100% Power Load Rejection Test (Unit 1, 1981)
2) 96% Power Feedwater Pump Trip Test (Unit 1, 1981)
3) 100% Power Water Level Setpoint Change (Unit 2, 1982)
4) 68% Power Recirculation Pump Trip Test (Unit 2, 1982)

On October 23, 1997 while Unit 1 of Kuosheng was operating at full power, the air supply nipple pipe break led to the F028A MSIV failed close. The increased pressure and power resulted in high steam flow in the other three steam lines, which in turn led to all MSIVs closure and the reactor scram. The event of a single MSIV closure reactor trip was analyzed[3] with the TITRAM codes and models. Figure 4 show good agreements between the plant record and the calculated results.

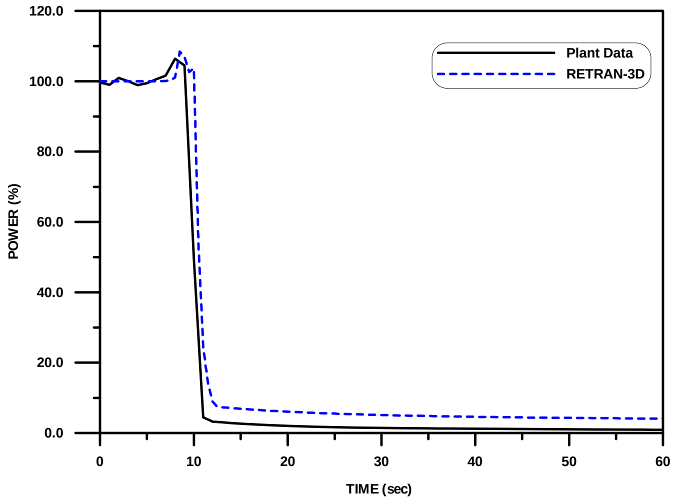

Figure 3 Single MSIV closure transient analysis

The Kuosheng Hot-Channel Model (HCM) with the associated fuel design and CPR correlation is also evaluated including the following three areas:

1) Steady state thermal hydraulic parameters: compared with vendor's analysis results,
2) Hot-channel flow rate: steady-state and transient results compared with vendor's analysis results
3) CPR correlation: steady state CPR results compared with measured dryout test data, and transient results compared with measured transient test data and vendor's analysis results.

The work of qualification through tests and full scale plant data to evaluate the capability of TITRAM code packages has been performed, reviewed and approved[4].

PHENOMENA IDENTIFICATION AND RANKING

The limiting events which set the operating limit minimum CPR (OLMCPR) were identified and analyzed as documented in the Reload Licensing Analysis[20][21] (RLA) for many of the previous and current reload cycles to ensure that the acceptance limits are not surpassed during these transients. The three significant transient events are FWCFNB, TTNB and LRNB.

The FWCFNB Event

An inadvertent failure of the feedwater control system causes the flow from the feedwater spargers to increase to the maximum achievable flow. As a result of the mismatch between the steam flow and the feedwater flow, the reactor vessel water level increases and the coolant temperature at the core inlet decreases. The increase in core inlet subcooling causes the power to increase. As the feedwater flow continues at the maximum demand, the water level increases until the high water level trip setpoint (L_8) is reached. Then there will be a reactor scram and a feedwater pump trip. After a delay from the L_8 signal, there will be a closure of the turbine stop valves (TSV) and a fast closure of the turbine control valves (TCV), which will cause a compression wave to travel through the steam lines into the vessel and cause a pressurization condition. Condenser bypass flow, which would mitigate the pressurization effect, is not credited in the analysis. The excursion of the core power due to void collapse is primarily terminated by reactor scram and void growth due to the recirculation pump high-to-low speed transfer. The recirculation pumps will transfer from high-to-low speed when the TSV reach 90% open.

The TTNB Event

The turbine trip causes a closure of all four TSVs. The resulting compression wave travels through the steam lines into the vessel and creates the rapid pressurization condition. A reactor scram and the recirculation pump high-to-low speed transfer are initiated when the TSV reach 90% open. Condenser bypass flow is not credited either. The excursion of the core power due to void collapse is primarily terminated by reactor scram, and void growth due to the recirculation pump high-to-low speed transfer.

The LRNB Event

The LRNB event is similar to the TTNB event except that LRNB is caused by a closure of the four TCVs rather than TSVs. Since the four TCVs have their own openings, the closure of TCVs might lead to more severe results even that the TCV has longer stroke time for a full opening TCV.

Phenomena Identification

These three events are pressurization events with similar scenarios. In the FWCFNB event, there are two phases with completely different phenomenology. In the first phase a power increase is produced by the core inlet subcooling increase. The second phase is marked by the system pressurization. The phenomena that significantly affect the CPR calculation in the transient are the following:

- Subcooling phase : The heat flux in this phase is determined by:
  1) Initial vessel level: this determines the length of the subcooling phase.
  2) Subcooling variation: the volume of liquid accumulated in the vessel affects the inlet subcooling, a lower initial liquid inventory causes a greater effect of the increasing subcooling feedwater flow.
  3) Neutronic void feedback when the core inlet subcooling increases.
  4) System initial values.

The core flow variation is not so important during this phase of the transient because the variation is very small and smooth due to mild changes of core power and system pressure.

- Pressurization phase : the heat flux in this phase is determined by:
  1) System pressure increase as consequence of the closure of the turbine stop valves.
  2) Positive void reactivity caused by pressurization effect.
  3) Insertion of negative reactivity due to scram.
  4) Gap conductivity.
  5) System initial values.

The core flow is affected by recirculation pump coastdown, pressurization waves in the vessel, and characteristics of the jet pumps.

The TTNB and LRNB events are pressurization transients, which are very similar to the pressurization phase in the FWCFNB event. This similarity between the events allows us to apply the same methodology to analyze the critical power ratio in these transients. Although they have similar phenomena, they are all significant events in determining the OLMCPR because under different operating power levels, one of them might set the limit.

With the key phenomena identified, it is possible to identify the significant input parameters that may have influence on ΔCPR. These are:

- Kinetic Feedback
  1) Cross sections and power profile generated with SIMULATE-3
  2) Coefficients of phase separation model
  3) Subcooled void option for the neutronic feedback

- Subcooling Variations
  4) Downcomer volume
  5) Lower plenum volume
  6) Temperature transport delay model

- Pressurization rate
  7) Separator pressure drop
  8) Steam line pressure drop
  9) Steam line volume
  10) Steam line inertia
  11) Recirculation loop volume
  12) Vessel dome volume
  13) Core pressure drop

14) Separators inertia
15) Dome with non-equilibrium pressurizer option
16) Calculation time steps

- Initial conditions
17) Kinetics data
18) Scram curve
19) Insertion of negative reactivity due to scram (rod worth)
20) Valve closure times and delays
21) Setpoint trips and delays
22) Power
23) Flow
24) Pressure
25) Level
26) Core inlet enthalpy
27) Gap conductivity
28) Power distribution

Many of these parameters are not independent and it is not always easy to identify the influence of each one on the safety criteria parameter. To determine the relevance of these variables and select the significant ones, we have taken into account expert opinions based on different references [6][23][25][26][27][23][23]. From the above sources we have obtained the most influential variables on the CPR as follows:

1) Steam line pressure drop
2) Steam line inertia
3) Jet pump M ratio
4) Steam line volume
5) Recirculation loop volume
6) Steam dome volume
7) Separators inertia
8) Downcomer volume
9) Lower plenum volume
10) Moderator direct heating
11) Phase separation models for KAPPA
12) Phase separation models for CGL
13) Temperature transport model
14) Scram speed
15) Initial level
16) Power profile (Axial Offset, *AO*)
17) Gap conductivity
18) Initial power
19) Rod worth
20) Core pressure drop
21) Separator Carryunder
22) Separator Pressure Drop
23) Initial Reactor Pressure

Discussion of the detailed phenomena of the above 23 parameters is given in the Topical Report[1]. The axial offset (*AO*) is defined as a fraction of the difference of power in the upper and lower half cores. Since there are 25 nodes axially of the fuel bundles in the reactor core simulated with RETRAN-3D and SIMULATE-3 models, the *AO* can be calculated as: ：

$$AO = \frac{\sum_{i=14}^{25} Q_{Node,i} - \sum_{i=1}^{12} Q_{Node,i}}{\sum_{i=1}^{25} Q_{Node,i}} \qquad (1)$$

where $Q_{Node,i}$ is the normalized node power for axial node *i* calculated by SIMULATE-3.

Ranking through Sensitivity Analysis

A Phenomena Identification and Ranking Table (PIRT) with an uncertainty value for each identified parameter to cover 95% of its possible values are established for the specific fast transients. Each parameter in the PIRT table has an uncertainty associated with a distribution. Usually instrumentation uncertainties such as power, pressure, temperature, and flow have two-side normal distributions *F(x)*. The uncertainty should bound plus and minus two times the standard deviation ($\pm 2\sigma$) yielding nearly 95% of the total area under the distribution. The determination of the 95% uncertainty value is based on data from the Principle Plant Parameters[28] of KSNPS, and the licensing analyses similar to that for Kuosheng by other experienced industry organizations[22][24][29].

Sensitivity studies based on the Analysis Nominal Model are performed to determine the ranking of the impact from uncertainties of identified parameters. A Licensing Model is defined by inserting conservative values for high ranking parameters.

UNCERTAINTY EVALUATION FOR CPR EVENTS

The main focus of uncertainty evaluation is to determine the impact on the primary safety parameters (ΔCPR or system pressure) by inputting the uncertainties identified in PIRT into the Analysis Nominal Model. Then, the individual uncertainties are combined into an overall uncertainty. Different statistical approaches[6] have successfully been used to determine the combined uncertainties. The statement of total uncertainty for TITRAM is given as a statement of probability for the limiting value of the primary safety criteria parameter. A Licensing Model is defined to give results which cover the overall uncertainty.

Analysis Nominal Model

The KSNPS “BASEDECK” model has been verified with a set of plant startup test and plant transient event analyses. A modeling approach similar to the PBTT analysis is adopted for the Kuosheng RETRAN-3D model and the accuracy of this model to simulate plant system behavior has been demonstrated[3]. The Nominal Model using nominal values such as nominal reactor power at 101.7% of original design rated power at 2894 MWt. is used to perform sensitivity studies of the selected fast transient analysis.

Determination of The Most Limiting Initial Conditions

In order to perform a useful sensitivity study, the most limiting initial conditions leading to the most severe impact on the safety criteria parameter have to be determined. In the rated power condition, the plant can be operated under MEOD, NOMINAL, or ICF conditions. MEOD means maximum extended operating domain where the core flow can be as low as 77% of rated. NOMINAL means 100% rated core flow, and ICF means increased core flow which can be as high as 105% of rated core flow. The transient ΔCPR is then calculated to determine the most severe case.

Sensitivity Analysis

The following sensitivity study steps are used:

1) Use the Nominal Model to calculate ΔCPR as the base case.
2) Fix all the inputs except the one for sensitivity study, use the uncertainty value which covers 95% of its possible value and calculate ΔCPR.
3) Continue to do so until all the parameters are done with the ΔCPR calculation.

Through each sensitivity case, an initial CPR (ICPR) is determined, and a ΔCPR can be calculated where ΔCPR is the difference between ICPR and the minimum CPR (MCPR) during the transient event.

$$\Delta CPR = DCPR = ICPR - MCPR \tag{2}$$

The ratio can be defined as:

$$RCPR = \frac{DCPR}{ICPR} \tag{3}$$

Then a case of sensitivity study can lead to results as:

$$DRCPR_i = RCPR_i - RCPR_{nominal} \tag{4}$$

Where i is the i'th sensitivity case result. In this way all sensitivity cases can be compared.

Statistical Treatment

A statistical approach by adding uncertainty on top of the nominal calculation result for ΔCPR is used. Since 95% uncertainty value is used to calculate RCPR for each parameter in the PIRT table, we can define an overall calculation uncertainty (based on all of the sensitivity cases) as:

$$DRCPR(95) = \sqrt{DRCPR_1^{\,2} + DRCPR_2^{\,2} + ... + DRCPR_n^{\,2}} \tag{5}$$

From Equation (4) *DRCPR(95)* can be rewritten as:

$$DRCPR(95) = RCPR(95) - RCPR_{norminal} \tag{6}$$

Equation (3) can be substituted into Equation (6) to solve for the unknown *RCPR(95)*:

$$DRCPR(95) = \frac{DCPR(95)}{ICPR_{95}} - RCPR_{nominal} \tag{7}$$

Rearranging Equation (7), we can get:

$$DCPR(95) = [DRCPR(95) + RCPR_{\text{nominal}}] \times ICPR_{95} \tag{8}$$

where $ICPR_{95}$ is defined as the average ICPR from n sensitivity cases:

$$ICPR_{95} = \frac{ICPR_1 + ICPR_2 + ... + ICPR_n}{n} \tag{9}$$

where n dose not include the base case. Therefore, based on the statistical approach, one can calculate a ΔCPR with 95% probability to cover overall uncertainties for a given transient and PIRT.

Results of Uncertainty Analysis

Based on the Nominal Model analysis for the three fast transients from the three different initial conditions, the case of ICF has been identified to be the most limiting initial condition for all of the three events. The FWCFNB at ICF gives the most limiting result of ΔCPR for among all nine cases. The sensitivity study starts from the base case of ICF for each of the three transients with nominal values listed in the PIRT table for the identified parameters. The ΔCPR is calculated based on TITRAM codes and models for the fuel. The base case ΔCPRs for FWCFNB is 0.077, for TTNB is 0.067, and for LRNB is 0.074.

Table 3 Results of Statistic Analysis of Overall Uncertainty

	FWCFNB	TTNB	LRNB
$\Delta CPR_{Nominal,\ ICF}$	0.077	0.067	0.074
$RCPR_{Nominal}$	0.068	0.059	0.065
DRCPR(95)	0.016	0.033	0.022
ICPR	1.15	1.14	1.14
DCPR(95)	0.096	0.11	0.095

Table 3 shows the overall uncertainty based on 95% uncertainty value for each parameter in the PIRT table. In order to evaluate the effect of each case, the significance of each parameter on CPR can be identified. A variable of significance (S_i) can be defined as:

$$S_i = \frac{DRCPR_i^{\,2}}{DRCPR_1^{\,2} + DRCPR_2^{\,2} + ... + DRCPR_n^{\,2}} \tag{10}$$

which can be used to see how each parameter contributes to the overall uncertainty. Three parameters are identified as the most influential parameters to the safety criteria parameter, namely, phase separation models for KAPPA, scram speed, and *AO* as listed in Table 4 for the case of TTNB.

Licensing Model Analysis

The Licensing Model is defined for the following condition:

$$DCPR(LM) > DCPR(95) \quad (11)$$

The licensing model input parameters are all the same as those in the Analysis Nominal Model except KAPP, scram speed, axial power profile and an assumption of safety relief values. Conservative values for the three parameters are used.

Table 4 Ranking of PIRT Parameters for TTNB

Rank	Parameters	*DRCPR*		S_i (%)
1.	KAPPA Algebraic slip model		0.02275	46.71
2.	Scram speed(95%)		0.02108	40.10
3.	AO Axial Power Distribution		0.01116	11.24
4.	Separators inertia		0.00276	0.69
5.	Moderator direct heating	-	0.00197	0.35
6.	Steam line volume		0.00188	0.32
7.	Steam dome volume		0.00157	0.22
8.	Steam line pressure drop		0.00144	0.19
9.	Rod worth(Limiting)		0.00075	0.05
10.	Jet pump M ratio	-	0.00074	0.05
11.	Core pressure drop		0.00048	0.02
12.	Initial Reactor Pressure	-	0.00039	0.01
13.	Steam line inertia	-	0.00038	0.01
14.	Initial level		0.00037	0.01
15.	CGL Algebraic slip model		0.0003	0.01
16.	Initial power	-	0.00029	0.01
17.	Gap conductivity	-	0.00027	0.01
18.	Downcomer volume		0.00015	0.00
19.	Separator Carryunder		0.00007	0.00
20.	Lower plenum volume		0.00004	0.00
21.	Temperature transport model(Yes)		0.00001	0.00
22.	Separator Pressure Drop	-	0.00001	0.00
23.	Recirculation loop volume		0	0.00

KAPPA Drift Flux Model Parameter : The KAPPA parameter is the geometric channel flow factor of the algebraic phase separation model with the correlation of Zollotar-Lellouche[13] drift flux. Modification of the value may lead to change of the void fraction used for the neutronic feedback. A 95% uncertainty value is used.

Scram Speed : Two sets of scram speeds are used in the Licensing Model. One is the Technical Specification scram speed as Option A. The other is the set of 95% uncertainty speed which is used as Option B.

AO of Axial Power Profile : The axial power profile is always a concern in a pressurization event. The sensitivity study shows that AO is significant to the final result of DCPR. This has been considered in the reload design by the fuel suppliers. A design basis step through is used for the plant operation and a licensing step through with more top peaked axial power shape at the end of cycle (higher AO) is used in safety analysis. Basically the AO from licensing basis step through could cover the design basis AO and associated uncertainty. In TITRAM Licensing Model of KSNPS, we use the licensing basis step through from SIMULATE-3 calculation to generate kinetics file TAPE40 which is able to cover the uncertainty effect of AO.

A statistical adder will be used to cover the uncertainty associated with the variables which are not introduced into the Licensing Model. It is determined by the root mean square method as :

$$Adder_{event} = \sqrt{\sum_{i=1}^{n} \mathrm{DRCPR}_i^{\,2} - \mathrm{DRCPR}_{\mathrm{KAPPA}}^{\,2} - \mathrm{DRCPR}_{\mathrm{scram95}}^{\,2} - \mathrm{DRCPR}_{\mathrm{AO}}^{\,2}} \quad (12)$$

Based on the Licensing Model with Technical Specification scram speed, we can calculate RCPR from TITRAM analysis as

$$RCPR_{OpA} = \frac{DCPR_{OpA}}{ICPR_{OpA}} \quad (13)$$

We can define the Licensing Model DCPR for Option A as:

$$DCPR_{LM,OpA} = \frac{SLMCPR}{1-\left[RCPR_{OpA} + Adder\right]} - SLMCPR \quad (14)$$

Based on the Licensing Model with 95% probability scram speed, we can calculate RCPR from TITRAM analysis as

$$RCPR_{OpB} = \frac{DCPR_{OpB}}{ICPR_{OpB}} \quad (15)$$

We can define the Licensing Model DCPR for Option B as:

$$DCPR_{LM,OpB} = \frac{SLMCPR}{1-\left[RCPR_{OpB} + Adder\right]} - SLMCPR \quad (16)$$

Results of Licensing Model Analysis

Table 5 shows the results of Licensing Model Analysis for the three selected pressurization events with Option A and Option B. Based on the CSAU approach of uncertainty evaluation, the overall uncertainty of TITRAM on KSNPS fast transients has been quantified. The conservatism has been added to the best estimated analysis to be Licensing Model analysis. It can be found that Option A is very conservative and Option B is quite reasonable. The difference between Option A and Option B results is 0.09 for FWCFNB, TTNB, and LRNB. Finally, LRNB has the highest Licensing Model DCPRs for both Option A and Option B.

Table 5 Results of Licensing Model Analysis

		Licensing Model Analysis				
		ICPR	DCPR	RCPR	Adder	Licensing Model DCPR
Option A	FWCFNB	1.26	0.19	0.15	0.0076	0.21
	TTNB	1.28	0.21	0.16	0.0046	0.22
	LRNB	1.28	0.21	0.17	0.0084	0.23
Option B	FWCFNB	1.18	0.11	0.09	0.0076	0.12
	TTNB	1.19	0.12	0.10	0.0046	0.13
	LRNB	1.19	0.12	0.10	0.0084	0.14

UNCERTAINTY EVALUATION FOR ΔP EVENTS

In addition to the thermal limit evaluation, the pressure limit analysis is also conducted every reload. The RLA[20][21] defined the analysis as the ASME overpressurization analysis. The analysis is performed to demonstrate compliance with the ASME Boiler and Pressure Vessel Codes. It also verifies that the safety valves have sufficient capacity and performance to prevent the pressure from reaching the established transient pressure safety limit, which is 110% of the design pressure.

The uncertainty evaluation for vessel overpressure limit follows the CSAU approach. The phenomena identification of the three transients, namely ASME-MSIVC, ASME-TSVC, and ASME-TCVC, is to see if the PIRT table established in thermal limit analysis is applicable or not for the uncertainty evaluation. Therefore the uncertainty evaluation can be conducted based on the previous PIRT table and one of the sensitivity studies with similar phenomena. The adder of peak pressure can then be determined.

The ASME Overpressurization Transient

The ASME overpressurization transient is initiated with closure of the MSIV, TSV, or TCV. This causes a pressure wave that spreads through the steam lines up to the vessel. The vessel output steam flow limitation makes the dome pressure increase. The vessel pressure increase produces a core void collapse increasing the moderator reactivity and producing a power excursion. The power increase ends with the negative reactivity insertion due to scram which is activated by high neutron flux signal rather than position scram from MSIV, or TSV, or TCV. Additionally, the recirculation pump trip is activated by reaching system high pressure setpoint.

Phenomena Comparing to TTNB's

It can be concluded that the phenomenology for the above three transients is almost identical to that of the TTNB. The power increase is due to vessel pressurization caused by closure of valves in the steam line with the loss of turbine and turbine bypass. In the same way, the power excursion is controlled by the fast insertion of the control rods and the vessel is depressurized by the opening of the safety-relief valves. Finally, the flow evolution through core is also similar, given that in these transients the trip or the low speed transfer of recirculation pumps is produced.

Sensitivity Analysis

The sensitivity analysis starts from the base case of TTNB Nominal Model analysis. For overpressurization events, the primary safety parameter is the maximum change of vessel pressure ($\Delta p_{vessel\ peak}$) as:

$$\Delta p_{vessel\ peak} = p_{vessel\ peak} - p_{lower\ plenum\ initial} \tag{17}$$

where $p_{vessel\ peak}$ is the calculated peak vessel pressure in TTNB transient and $p_{lower\ plenum\ initial}$ is the initial lower plenum pressure. The sensitivity study is carried out to get the change of $\Delta p_{vessel\ peak}$ ($D\Delta p_{vessel\ peak}$) as:

$$D\Delta p_{vessel\ peak,i} = \Delta p_{vessel\ peak,i} - \Delta p_{vessel\ peak,base} \tag{18}$$

The 95% probability of $D\Delta p_{vessel\ peak}$ can be calculated as:

$$D\Delta p_{vessel\ peak\ 95} = \sqrt{D\Delta p^2{}_{vessel\ peak,1} + D\Delta p^2{}_{vessel\ peak,2} + \cdots + D\Delta p^2{}_{vessel\ peak,n}} \tag{19}$$

Table 6 shows the results of uncertainty evaluation of TITRAM pressure limit analysis. The final $D\Delta p_{vessel\ peak\ 95}$ is 11.6 psi which is very close to the value obtained by an experienced analysis team[22].

Table 6 Peak Pressure Uncertainty Evaluation of TTNB

No.	Parameters	$\Delta p_{vessel\ peak}$	$D\Delta p_{vessel\ peak}$
	Base	141.89	0.00
1	Steam line pressure drop	139.76	-2.13
2	Steam line inertia	140.41	-1.48
3	Jet pump M ratio	139.11	-2.78
4	Steam line volume	139.20	-2.69
5	Recirculation loop volume	138.91	-2.98
6	Steam dome volume	140.77	-1.12
7	Separators inertia	139.05	-2.84
8	Downcomer volume	138.92	-2.97
9	Lower plenum volume	138.92	-2.97
10	Moderator direct heating	139.12	-2.77
11	KAPPA Algebraic slip model	143.65	+1.76
12	CGL Algebraic slip model	138.93	-2.96
13	Temperature transport model(Yes)	138.90	-2.99
14	Scram speed(95%)	142.26	+0.37
15	Initial level	139.29	-2.60
16	AO Axial Power Distribution	141.58	-0.31
17	Gap conductivity	140.61	-1.28
18	Initial power	139.31	-2.58
19	Rod worth(Limiting)	139.32	-2.57
20	Core pressure drop	138.95	-2.94
21	Separator Carryunder	138.90	-2.99
22	Separator Pressure Drop	138.89	-3.00
23	Initial Reactor Pressure	141.74	-0.15

Licensing Model Analysis

Therefore, given that the phenomenology is similar to that of the TTNB, the categorization of the variables and demonstration of Licensing Model conservatism method carried out for the TTNB is valid for the three ASME transients. A similar Licensing Model to the one defined in the TTNB analysis is able to be used for the ASME overpressurization transients. The adder to the vessel pressure obtained in the TTNB can also be applied to the three ASME transients.

A Licensing Model analysis can be performed for the three ASME overpressurization transients. The results of peak vessel pressure with adder are compared with the pressure safety limit of 1390 psia to demonstrate compliance with the ASME Boiler and Pressure Vessel Code. The vessel peak pressure of Licensing Model analysis is therefore determined as

$$p_{vessel\ peak,LM} = p_{vessel\ peak,LM\ calculated} + \mathrm{D}\Delta p_{vessel\ peak}95 \qquad (20)$$

For ASME overpressurization analysis, the Licensing Model vessel peak based on the ASME-TSVC transient is：

$$p_{vessel\ peak,LM} = 1280 + 11.6 = 1291.6 < p_{vessel\ design\ safety\ \mathrm{limit}} \qquad (21)$$

CONCLUSIONS

This paper presents the TITRAM fast transient analysis methodology for KSNPS. The purpose of this work is to provide a technical basis of qualification to perform plant specific licensing safety analyses for the FSAR system fast transients, and related plant operational transient analyses for the Kuosheng plant. The major task is to adequately quantify the TITRAM methodology uncertainty.

The TITRAM methodology adopts key CSAU concepts by using the important segments, namely code applicability, transient and phenomena identification and ranking, and uncertainty evaluation. A PIRT table with uncertainty values for each identified parameter to cover 95% of its possible values is established for the selected fast transients. The previously approved methodologies conducted by experienced organizations are used as references of the PIRT table as illustrated.

Sensitivity studies and statistical analyses are performed to determine the overall uncertainty of TITRAM based on the Analysis Nominal Model. A Licensing Model is defined by demonstrating that the safety limit parameter (such as ΔCPR) from the Licensing Model analysis conservatively bounds the Nominal Model result combined with the overall TITRAM uncertainty. Finally, the Licensing Model for KSNPS is established for future licensing applications.

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