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CHINSHAN LIVING PRA MODEL USING NUPRA SOFTWARE PACKAGE

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ABSTRACT

A living probabilistic risk assessment (PRA) model has been established for Chinshan Nuclear Power Station (BWR-4, MARK-I) using NUPRA software package. The core damage frequency due to internal events, seismic events and typhoons are evaluated in this model. The methodology and results considering the recent implementation of the 5th emergency diesel generator and automatic boron injection function are presented. The dominant sequences of this PRA model are discussed, and some possible applications of this living model are proposed.

I. INTRODUCTION

The probabilistic risk assessment (PRA) project of Chinshan Nuclear Power Station (CNPS) was started in 1988 and completed in 1991. In this original Chinshan PRA, the internal events and external events of earthquake, typhoon, fire and flood were analyzed. The total core damage frequency (CDF) is 1.4×10^{-4} per reactor-year(r-y), in which the contribution distributions are: internal event, 44%; seismic event, 45%; fire, 10%; typhoon, 0.2%; and flood, 0.1% (Ref. 1).

At the time that this model was constructed, the software for living PRA was rapidly developing. Also, the important design changes for CNPS, i.e., installation of the 5th emergency diesel generator (EDG; CNPS has two units, each unit had two EDGs originally) and the implementation of the redundant reactivity control system (RRCS; mainly adding automatic boron injection function in standby liquid control system), have been completed recently. For revising the PRA model and future application of this model, the living Chinshan PRA model was established in a personal computer using the latest version of NUPRA (2.1) software[2]. A methodology is developed for including the events induced by earthquake and typhoon in the living PRA model. However, due to the different methodology used for fire and flood analyses, there events are currently not included in this model.

In this paper, the Chinshan living PRA model and results are presented. Dominant core damage sequences from internal events and seismic events are discussed. Finally, the possible applications of this living model are proposed.

II. LIVING PRA MODEL AND RESULTS

II.A. Internal Events

Fifteen initiating events are identified in the Chinshan PRA. The event category and estimated annual frequency for these events are described in Table 1. Event tree is developed for each initiating event to delineate the accident sequences. The last item of this table, anticipated transient without scram (ATWS), is not an initiating event, but event trees are specifically developed for analyzing these consequential events.

In preparing the living model, the fault tree are modularized for saving memory and (Boolean) operation. Fault tree linkage method, i.e., small event tree large fault tree, is used for the event tree model and sequence quantification. In the sequence analysis, the same function (for the headings in the event tree) might be assigned differently under different situations. The detail of these processes has been presented in a previous PSA meeting[3].

The results of CDF initiated from each initiating event are also presented in the last column of Table 1; the total CDF for internal events is estimated to be 2.4×10^{-4} per r-y. The value in the last row of this table is the sum of all sequences due to ATWS. Due to the installation of the 5th EDG, ninety five percent

Table 1. Internal Initiating Events for Chinshan PRA

Event Category	Initiating Frequency (1/r-y)	CDF (1/r-y)
T_{1A}: Reactor isolated, feedwater available	0.69	$4.8x10^{-8}$
T_{1B}: Reactor isolated, feedwater trip	1.7	$2.1x10^{-6}$
T_{2A}: Reactor not isolated, feedwater available	3.8	$1.2x10^{-7}$
T_{2B}: Reactor not isolated, feedwater trip	1.1	$1.2x10^{-6}$
T_3: Loss of offsite power	0.15	$9.5x10^{-6}$
T_4: Inadvertent stuck open relieve valve	$8.0x10^{-2}$	$3.4x10^{-8}$
T_5: Loss of feedwater	$3.0x10^{-2}$	$9.5x10^{-7}$
T_C: Loss of compressed air	$1.5x10^{-3}$	$3.4x10^{-6}$
T_S: Loss of CSCW	$6.4x10^{-4}$	$2.9x10^{-7}$
A: Large LOCA	$2.7x10^{-4}$	$3.0x10^{-7}$
S_1: Intermediate LOCA	$2.7x10^{-3}$	$6.5x10^{-7}$
S_2: Small LOCA	$2.7x10^{-2}$	$< 10^{-8}$
A_0: LOCA bypassing containment	$8.4x10^{-6}$	$1.1x10^{-7}$
V_A: Interfacing LOCA	$4.5x10^{-6}$	$2.4x10^{-7}$
R_V: Vessel rupture	$2.7x10^{-7}$	$2.7x10^{-7}$
C_M: Anticipated transient without scram	$1.4x10^{-4}$*	$4.5x10^{-6}$

* (All transient initiating event frequency, 7.6) x (Failure of rod insertion per demand, $1.8x10^{-5}$)

of the CDF distributes about evenly in five transient scenarios: loss of offsite power (20%, excluding station blackout), transient without injection (20%), loss of long-term heat removal (19%), ATWS (19%), and station blackout (17%), and five percent from LOCA sequences.

II.B. Seismic and Typhoon Events

The seismic fragilities for about one hundred important equipment/structures in Chinshan Unit 1 are evaluated for Chinshan PRA. However, only eighteen of them are found to have a median peak ground acceleration (PGA) < 3g or a high-confidence-low -probability-of-failure (HCLPF) acceleration < 1g. Table 2 depicts these equipment/structures and their fragilities. Due to the strong fragility of the other safety-related equipment/structures, the probability of annual seismically induced failure for those equipment /structures are negligible.

In order to incorporate the seismic (and typhoon) events into the living model, a methodology is developed to separate the seismic and non seismic (random) failures. This is possible because of the way the impact of the seismic failures is modeled. In a seismic event, all redundant components affected by a seismic failure are assumed to fail at the same time. Thus, there are very few mixed cut sets of the seismic* random type. One notable exception is failure of the condensate storage tank, which has to be ANDed with failure to switch over to the suppression pool as a failure mechanism for high pressure core injection (HPCI) and reactor core isolation cooling (RCIC). Because of the relatively high unavailability on demand of HPCI and RCIC from other causes, it is slightly non conservative, but not very significantly so, to omit this potential failure mechanism. These approximations allow the almost complete separation of the seismic and non seismic failure modes of systems.

This approach simplifies the quantification of seismic-event trees, because now the seismic parts of the quantification, which are done on SEISMIC code[4], can be separated from the non seismic (or random) parts, which are done on NUPRA, as in the case of the internal-event tree. The seismic-event analysis is started, therefore, with a front-end seismic-event tree, shown in Fig. 1, which models seismically induced failures of components and structures and defines the initial seismic-damage states. For those damage states not directly leading to core damage, a system event tree is developed, and which includes only non seismic failure.

The front-end seismic-event tree is evolved from a preliminary front-end seismic-event tree that included more seismic failures, specifically, an event representing seismically induced diesel generator

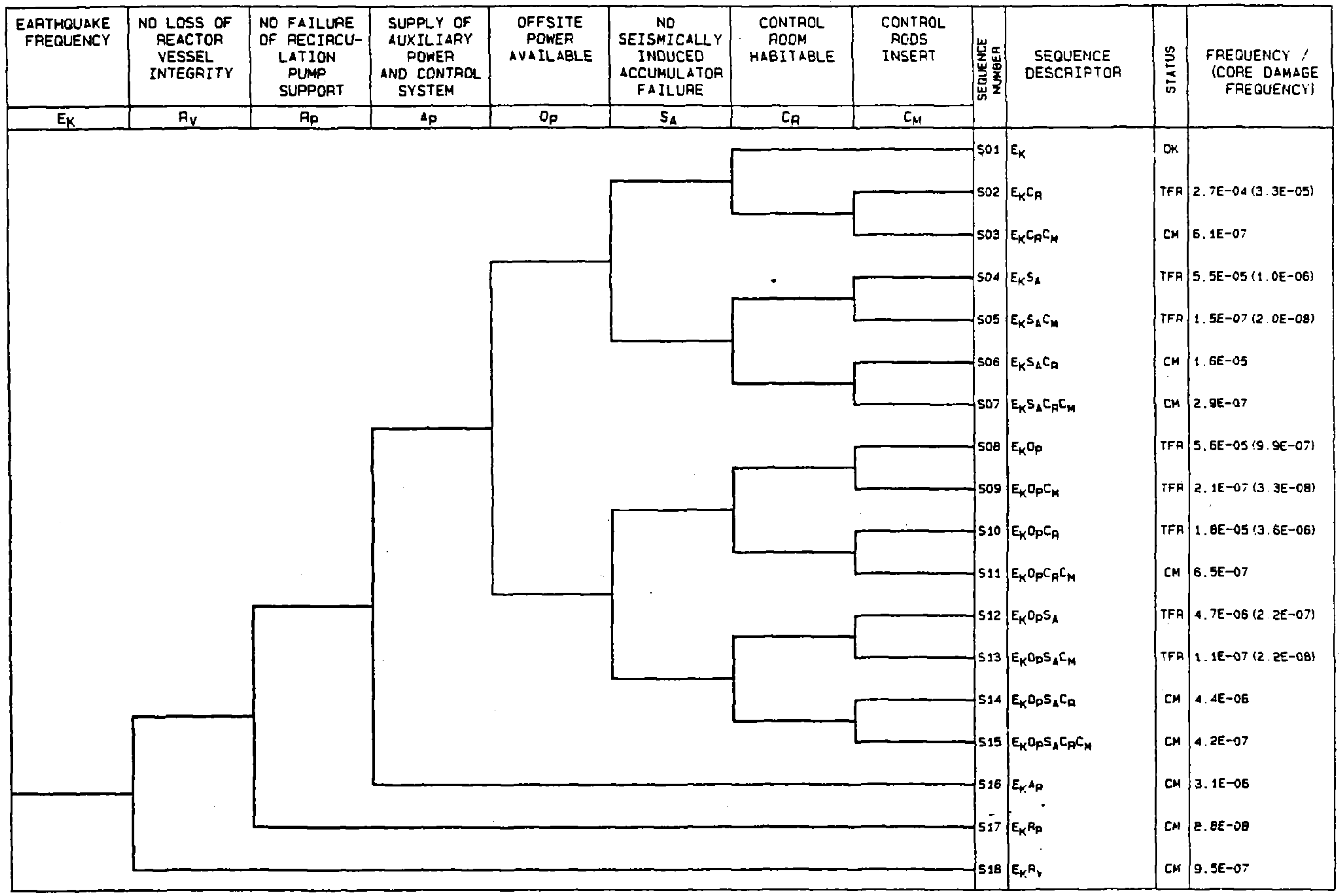

Figure 1. Front-end Seismic-event Tree

Table 2. Significant Seismically-induced Failures

Structure / equipment	PGA(g)	HCLPF(g)	Related Heading (Fig.1)
Control room ceiling	0.35	0.11	C_R
Offsite power (ceramic insulator)	0.40	0.19	O_P
Gas turbine	0.61	0.40	O_P
Air accumulator of relief valve	0.79	0.19	S_A
Power center transformers	1.03	0.31	A_P
Core support structure	1.05	0.29	C_M
Gas turbine motor control center	1.08	0.34	O_P
Conednsate storage tank	1.15	0.29	*
Reactor vessel support skirt	1.22	0.43	R_V
Switchgear (structure)	1.56	0.50	A_P
Fuel assembly	1.70	0.56	C_M
Recirculation pumps	1.77	0.59	R_P
Diesel oil storage tank	2.10	0.62	*
Switchgear (relay chatter)	2.25	0.80	*
Essensial service warer MCC	2.65	0.77	*
Main control boards	2.65	0.79	A_P
D/G control panel (chatter)	4.44	0.85	*
480-V power center (chatter)	5.66	0.88	*

* Negligible contributors to CDF and not modeled.

failures, and an event representing failure of all the electrically powered systems due to relay chatter. Solution of this original tree, however, shows that these failures are insignificant contributors to core damage frequency, hence they were left off the final tree used in the analysis.

The equipment/structures considered in each heading in the front-end tree are shown in the last column of Table 2. It is noteworthy that for control room habitable heading C_R, additional event of "loss of control room habitability given seismically-induced ceiling failure" is considered. A pseudo fragility for this event is assumed to be the same as that of control room ceiling. Furthermore, the fragility of offsite power in Table 2 is specifically for 345-kv transmission line ceramic insulator. Due to its different design, the fragility of 69-kv transmission line, to which the two gas turbines are connected, is considered to be very strong.

Out of eighteen plant damage conditions identified in Fig. 1, system event trees are developed for eight of them to further identify the accident sequences. The quantification of these event trees is performed by NUPRA software. The results of core damage frequencies initiated by these events are illustrated in the rightest column (values in parenthesis) of Fig. 1. The total seismically-induced CDF is estimated to be 6.6 x 10^{-5} per r-y.

Similar methodology is applied to evaluate the typhoon event induced core damage frequency. The strong wind induced loss of offsite power (due to failure of switchyard, or both transmission line and gas turbine building) frequency is estimated to be 1.1 x 10^{-5} per year. Using the same loss of offsite power event tree as that for seismically-induced loss of offsite power, the CDF is evaluated to be 1.2×10^{-7} per r-y. The original typhoon event induced CDF is 3.2×10^{-7} per r-y.

II.C. Safety Enhancement of Recent Design Improvements

In the orginal Chinshan PRA (before design improvements), the CDF for internals events was estimated to be 6.1×10^{-5} per r-y. Installation of the 5th EDG gives a CDF reduction of 3.6×10^{-5} per r-y. The implementation of the RRCS will prevent the operator error in failure to inject boron in time under ATWS. However, the dominant events for the ATWS sequence are valves set in wrong configuration in the standby liquid control system. This design change gives 5.0×10^{-7} per r-y reduction of CDF, about 10% of all ATWS sequences. It is noteworthy that in addition to failure to inject born, the operator errors in inhibiting automatic-depressurization system (ADS) and preventing vessel overfill are also evaluated in the

PRA model. These operator errors are found to be more crucial than initiating boron injection.

In the original Chinshan PRA, the seismically-induced CDF was estimated to be 6.3×10^{-5} per r-y. An error for the equation used for loss of long-term heat removal function in $E_K C_R$ (sequence S02 in Fig. 1) system event tree was found while performing the living PRA. After the correction, the CDF for Chinshan unit 1 (before design improvements) becomes 6.8×10^{-5} per r-y. Installation of the 5th EDG results in a 2.0×10^{-6} per r-y reduction of CDF, while the RRCS gives negligible improvement of seismically-induced CDF.

III. DISCUSSION OF DOMINANT SEQUENCES

III.A. Internal Events

In the original Chinshan PRA, the two most dominant sequences are both station blackout: one with failure to recover power in 10 hr (CDF = 3.6×10^{-5} per r-y), and the other with a stuck open relief valve and failure to recover power in 30 min. (CDF = 3.7×10^{-6} per r-y). After the installation of the 5th EDG, the CDF for these two sequences become 3.4×10^{-6} per r-y and 4.0×10^{-7} per r-y, respectively. The most dominant sequences changes to loss of offsite power and subsequent failure of all injection (not due to complete loss of ac power), which increases from 1.1×10^{-6} to 4.0×10^{-6} per r-y. This is because that before the installation of the 5th EDG, loss of combined structure cooling water (CSCW) will lead to failure of both original EDGs and all emergency core cooling systems (due to loss of room cooling for turbine-driven HPCI and RCIC, and loss of motor cooling for low pressure injection systems), and these events are categorized as cut sets in the station blackout sequence. Since the 5th EDG is independent of CSCW, after the design change these events will not lead to station blackout but will cause loss of all injection under loss of offsite power situation. The long-term station blackout (filure to recover power in 10 hr) sequence is however, still the second dominant sequence.

The third dominant sequence is initiated by loss of compressed air followed by loss of residual heat removal (RHR) system. Since the power conversion system and containment venting are not available under no compressed air circumstance, the long-term heat removal function is solely relied on the RHR system. This sequence has an estimated CDF of 3.3×10^{-6} per r-y, and the valve problems are the dominant contributors to this sequence. Considering the long period of time available before core damage, the recovery of these valve problems is highly likely, but currently this credit is not taken in the PRA model.

III.B. External Events

Due to the weak fragility of control room ceiling and the assumption of control room habitability described above, seismically induced loss of control room is the main initiator for core damage. Under seismically induced loss of control room situation, there is possibility that logic cabinets generating automatic safety system actuating signal will also be damaged. The most dominant sequence in seismic events is that auto-initiation of injection systems capability is lost (with assumed probability of 0.10), and the operators are unable to regain control from the alternate shutdown panel (ASP, with assumed probability of 0.7, mainly due to the reason that only in the office hour regain of control is possible by operators outside the main control room; operator in the main control room are likely to be striked by the falling ceiling). The frequency for this sequence is estimated to be 1.9×10^{-5} per r-y.

The second dominant sequence is seismically induced safety relief valve (SRV) accumulators failure and loss of control room, which has an estimated frequency of 1.6×10^{-5} per r-y. The assumed consequence of accumulators failure, i.e., loss of instrument nitrogen, will lead to SRVs unavailable for depressurization. Although the RCIC system can be operated from the ASP, its room cooling cannot. Thus, eventually RCIC fails due to no room cooling. Since the vessel can not be depressurized, the low pressure systems are unable to make up inventory. The above two sequences contribute more than 50% of the seismically-induced CDF.

A sensitivity study was performed to understand what CDF reduction can be obtained if the fragilities for control room ceiling and/or SRV accumulators are improved to very high values. Assuming no control room ceiling failure in a seismic event, the seismically induced CDF becomes 7.4×10^{-6} per r-y i.e., an 89% reduction. If no SRV accumulators failure is further assumed, the CDF decreases to 5.6×10^{-6} per r-y.

IV. CONCLUSIONS AND RECOMMENDATIONS

A living PRA for Chinshan Nuclear Power Station Unit 1 has been established using NUPRA software package. This model evaluates core damage sequences initiated by internal events, seismic events, and typhoons. A specific methodology decoupling the seismic failures and random failures is developed for the seismic event and typhoon analysis. The recently two design changes, i.e., installation of the 5th EDG and implementation of RRCS have been incorporated into the model. The first improvement gives a safety enhancement of $3.8x10^{-5}$ per r-y CDF reduction, while the second gives $5.0x10^{-7}$ per r-y CDF reduction. The CDF induced by internal events is estimated be $2.4x10^{-5}$ per r-y, by seismic events, $6.6x10^{-5}$ per r-y, and by typhoons, $1.2x10^{-7}$ per r-y. Considering CDF induced by fire events, $1.4x10^{-5}$ per r-y, currently the total CDF for Chinshan Unit 1 is $1.0x10^{-4}$ per r-y.

The results of seismic event analysis show that a very effective design modification for safety enhancement is the improvement of control room ceiling fragility. The seismically-induced CDF will be reduced by a factor of ten (89% reduction) after this modification. The Chinshan Plant has scheduled to accomplish this improvement in the near future.

This living PRA model has been transferred to the site, and the application of PRA will be performed by the plant engineers using this tool. It is planned to used this model to evaluate the performance indicator of plant safety based on the plant configuration. The safety enhancement of a design change requests will also be assessed by this model. Furthermore, this model can be used as a basis for the optimization of plant specific technical specification. On the other hand, this living model will be updated at least after every refueling outage to be consistent with the design chang, component failure experience, procedure and/or operator training improvement, and new knowledge about severe accident. In this way, the model grows with the plant, and is always a correct and handy tool for PRA application.

REFERENCES

1. " Probabilistic Risk Assessment for Chinshan Nuclear Power Station Unit 1, " Atomic Energy Council, Executive Yuan, Republic of China (1991).

2."NUPRA 2.1 User Manual," HALLIBURTON NUS Environmental Corporation (1993).

3. T.J. LIN, G.D. LI, and S..K. CHENG, " Insights Gained from Chinshan Living PRA Model in Taiwan," *Proc. PSA'93 Int. Topl. Mtg. Probabilistic Safety Assessment,* Clearwater Beach, Florida, January 26-29, 1993.

4. R. KARIMI, "SEISMIC- A Computer Program for Seismic Risk Evaluation," NUS-4364, NUS Cooporation (1983).