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Stress analysis of NPP steam generator and main factors affecting safety

核电站蒸汽发生器应力分析及影响安全的主要因素

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Abstract - **The steam generator (SG) serves as the primary means of removing the heat generated within the reactor core and is part of the reactor coolant system (RCS) pressure boundary in nuclear power plant (NPP). SGs are heat exchangers used to convert water into steam by the heat produced in the nuclear reactor core and deliver the steam to drive turbines to generate electricity. The main components of SGs are Equipment Safety Class 1 or Class 2, Seismic Design Category I and ASME B&PVC III NB components, whose structure integrity affects the safety of NPP directly. The stress analysis of SGs is performed under various loads by finite element models, such as seismic analysis, fatigue analysis, fracture analysis, flow-induced vibration (FIV) analysis, wear analysis, and so on. The results demonstrate that the stress combination and evaluation under each service level meet the requirement of the ASME B&PVC III.**

In addition, the main factors affecting structure integrity and safe operation of SGs are summarized and proposed with the weak parts of SGs. For example, the environmental temperature of various conditions affects the fracture analysis results significantly. The seismic analysis results are sensitive to the stiffness of SGs' supports and the structure setting of anti-vibration bars (AVBs). Moreover, more than ten thousands of tubes, which are the key and weak components and are located between the primary and secondary coolant loops of NPP, are susceptible to flow induced vibration, wear, corrosion and seismic damage. These comments are expected to be significant for future analysis and design.

Keywords – **nuclear power plant, steam generator, stress analysis, safety**

摘要 – 蒸汽发生器是排出反应堆堆芯热量的主要设备，是核电站反应堆冷却剂系统压力边界的一部分。蒸汽发生器通过传热管将反应堆冷却剂从堆芯获得的热量传递给二次侧介质转化为蒸汽，送入汽轮机组发电。蒸汽发生器的主要部件为核安全 1 级或 2 级、抗震 I 类、规范等级 ASME B&PVC III NB 部件，其结构完整性直接影响核电站的安全性。本文对蒸汽发生器通过有限元建模方式对其进行了各种载荷下的应力分析，包括抗震分析、疲劳分析、断裂分析、流致振动和磨损分析等。分析结果表明，各种使用限制下蒸汽发生器的应力组合和评定结果满足 ASME B&PVC Ⅲ的相关要求。

同时，本文归纳总结了可能影响蒸汽发生器结构完整性和安全运行的主要因素和结构薄弱部位。例如各种工况下的环境温度对断裂分析结果影响较大，设备支撑刚度和抗振条设置对地震反应结果影响较大。此外，蒸汽发生器中的万余根传热管处于一二回路边界，既是设备关键部件也是薄弱部件，易受流致振动、磨损、腐蚀、地震等影响而损坏。这些结论对后续蒸汽发生器的分析和设计具有一定的指导意义。

关键词 – 核电站，蒸汽发生器，应力分析，安全

I. INTRODUCTION

Nuclear power plant (NPP) is the thermal power station in which the heat source is a nuclear reactor. As is typical in all conventional thermal power stations, the heat is used to generate steam which drives a steam turbine connected to a generator which produces electricity. Nuclear power is an important source of energy. As of 16 February 2014, the IAEA report there are 438 nuclear power reactors in operation operating in 31 countries [1]. Specially, about 75% country's electricity production is from nuclear power in France which is the highest nuclear power percentage country in the world. And about 50% new NPPs are being built in China now.

On the other hand, operating nuclear reactors contain large amounts of radioactive fission products which, if dispersed, may pose a direct radiation hazard, contaminate soil and vegetation, and be ingested by humans and animals. Human exposure at high enough levels can cause both short-term illness and death and longer-term death by cancer and other diseases. Some famous accidents at nuclear power plants were the 2011 Fukushima nuclear disaster in Japan, 1986 Chernobyl disaster in Ukraine, and the 1979 Three Mile Island accident in the United States. Safety is one of the most important issues for Nuclear Power Plant, especially in post-Fukushima ages. The radioelement should be always kept in the Reactor Coolant System (RCS) safely.

Steam generators are heat exchangers used to convert water into steam by the heat produced in the nuclear reactor core. They are used in pressurized water reactors (PWR) between the primary and secondary coolant loops. The water flowing through the primary side of steam generator boils water in the shell side to produce steam in the secondary loop that is delivered to the turbines to make electricity. And steam generator is one of longest period production and heaviest equipments in NPP. The main components of SGs are Equipment Safety Class 1 or Class 2, Seismic Design Category I and ASME B&PVC III [2] NB components. Structure integrity of steam generator must be ensured to ensure the safety of NPP.

In this paper, stress analysis for steam generator of an abuilding nuclear power plant is performed to ensure the 60 years lifetime of the component with the FEM model. For example, seismic analysis, fatigue analysis, fracture analysis, flow-induced vibration (FIV) analysis and wear analysis are performed. And the analysis results of SG demonstrate that the stress combination and evaluation under each service level meet the requirement of the ASME B&PVC III [2]. In addition, the main factors affecting structure integrity and safe operation of SGs are summarized and proposed with the weak parts of SGs. For example, the environmental temperature of various conditions affects the fracture analysis results significantly. The seismic analysis results are sensitive to the stiffness of SGs' supports and the structure setting of anti-vibration bars (AVBs). Moreover, more than ten thousands of tubes, which are the key and weak components and are located between the primary and secondary coolant loops of NPP, are susceptible to flow induced vibration, wear, corrosion and seismic damage. These comments are expected to be significant for future analysis and design.

II. Geometry and Material of Steam Generator

The Steam Generator, shown in Fig.1, is a vertical shell and U-tube evaporator with integral moisture separating equipment. Its basic function is to transfer heat from the single-phase reactor coolant water through the U-shaped heat exchanger tubes to the boiling, two-phase steam mixture in the secondary side. The steam generator separates dry and saturated steam from the boiling mixture, and delivers the steam to a nozzle from which it flows to the turbine. In addition to its steady-state heat transfer function, the steam generator secondary side provides a water inventory which is available as a heat sink to mitigate primary side high temperature transients and to accommodate accident conditions. The principal subcomponents of the steam generator include the channel head, tubesheet, lower shell, tube bundle, steam drum, and moisture separating equipment. The channel head, tubesheet, and tube bundle form the primary pressure boundary for the reactor coolant.

The material of the main subcomponents is SA-508, Grade 3 Class 2 per ASME code, such as channel head, tubesheet, lower shell, upper shell and so on. The material of tube is Alloy 690. The material strengths and physical properties can be obtained from the ASME Code [3].

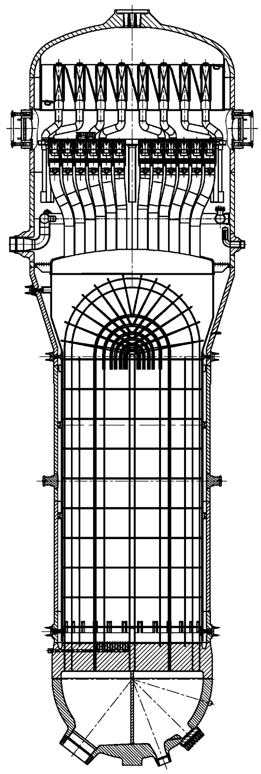


Fig.1, Structure drawing of Steam Generator

III. Loads Requirement of Analysis

The design basis of SG is sixty (60) years with 90 percent availability over the design life. Design temperature and design pressure of primary side is 350°C and 17.2 MPa, respectively. And design temperature and design pressure of secondary side is 320°C and 8.2 MPa. The steam generator shall be capable of satisfying functional and safety requirements under Design, Level A, Level B, Level C, and Level D conditions. The number of transient occurrences is based on a plant design life of 60 years. And the effects of pre-operational passive core cooling test transients shall be evaluated. The steam generator (including nozzles and supports) shall be analyzed using the load combinations. For example, internal design pressure, dead weight, Service Level A's design mechanical loads and external mechanical loads, such as the nozzle reactions associated with piping systems, are considered in design condition. And dynamic load associated with Level A (Normal) service conditions, peak pressure, dead weight, and external mechanical loads are considered in Level A condition. Especially, the load of Safe Shutdown Earthquake (SSE), postulated pipe rupture events and locked rotor is considered in Level D (Faulted) condition.

IV. ANALYSIS AND EVALUATION FOR STEAM GENERATOR

4.1 ANALYSIS AND CALCULATION

The analysis employs finite element methods using the ANSYS Computer Code for stress analysis mainly. The stress analysis of SGs is performed under various loads by finite element models, such as seismic analysis, fatigue analysis, fracture analysis, flow-induced vibration (FIV) analysis, wear analysis, and so on. The calculated stresses are compared to the allowable stress values calculated using S_m, S_y, and S_u values.

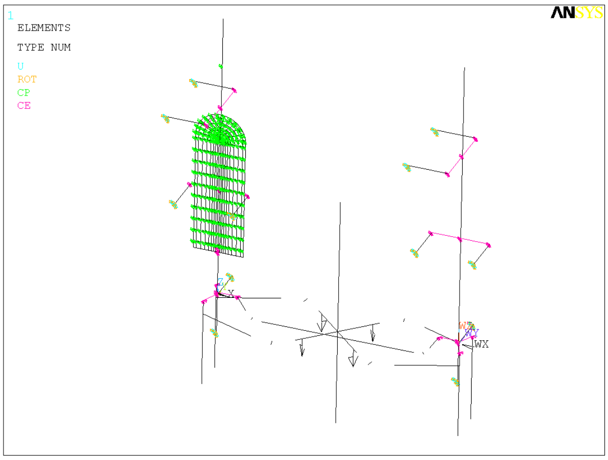


Fig.2, Seismic Model of SG with Reactor Coolant Loop (RCL)

The analysis model of SG is created with moisture separator assembly and tube bundle assembly. And the seismic analysis is performed with reactor coolant system RCS model with superelement element MATRIX50 under Safe Shutdown Earthquake. The seismic model of SG is shown in Fig.2 and Fig.3. The major structural components idealized in the finite element model include the steam generator shell (primary and secondary side pressure boundary), primary and secondary separators, wrapper, stayrods, deck plates, tube bundle, AVBs and tube support plates (TSPs). The model of the SG consists of a system of pipe elements PIPE16, beam elements BEAM4, and general matrix elements MATRIX27 (with both mass and stiffness options). The stiffness values of the lower, intermediate and upper supports are represented using one spring elements (COMBIN14). In Fig.2 and Fig.3, U is displacement constraints, ROT is rotation constraints. And CP defines a set of coupled degrees of freedom. CE defines a constraint equation relating degrees of freedom. The largest (purple) arrows mean the displacement constraints of the reactor pressure vessel support. The working plane shown by WX, WY, and WZ is an imaginary plane as the temporary modeling tool.

Moreover, LOCA Rarefaction Analysis and LOCA Shaking Analysis also apply this SG model without RCS model.

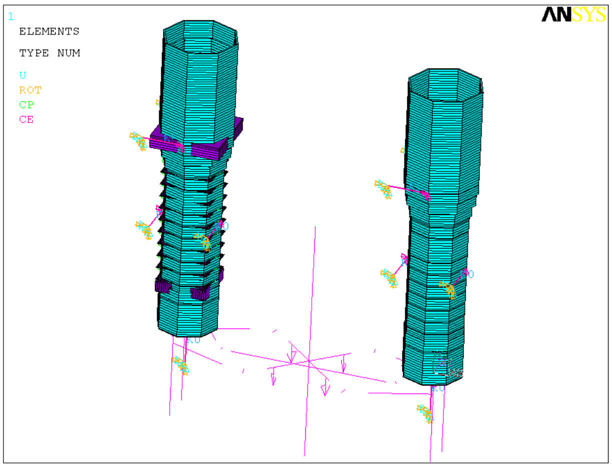


Fig.3, Seismic Model of SG (Display with elements shapes determined from the real constants [4])

Flow induced vibration analysis are perform to prevent the fluid-elastic instability (FEI) and obtains the vibration response of tubes, which is the input of wear analysis of SG. In wear analysis of SG tube, the Archard method is applied. To predict and evaluate fretting damage, the flow field in the equipment and flow-induced vibration response of SG tube is calculated and obtained. The Archard equation [5-6] is used to predict the wear as shown below.

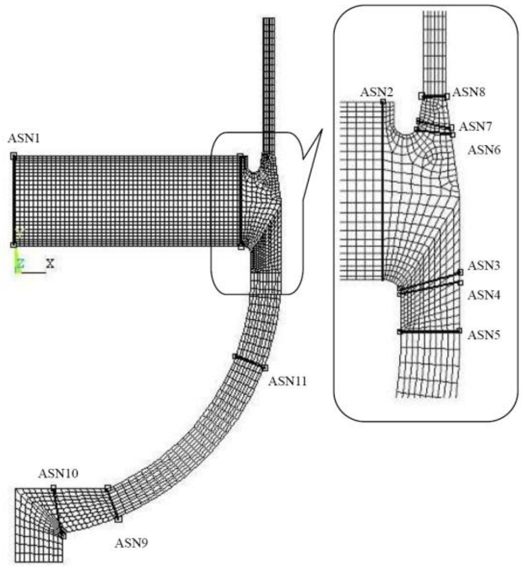
$$\dot{V} = K\dot{W} \tag{1}$$

where $\dot{V}$ is wear volume rate, $\dot{W}$ is work rate, and K is wear coefficient to be measured by experiment. Work rate is defined as the normal component of contact force, F, integrated over the real sliding distance s

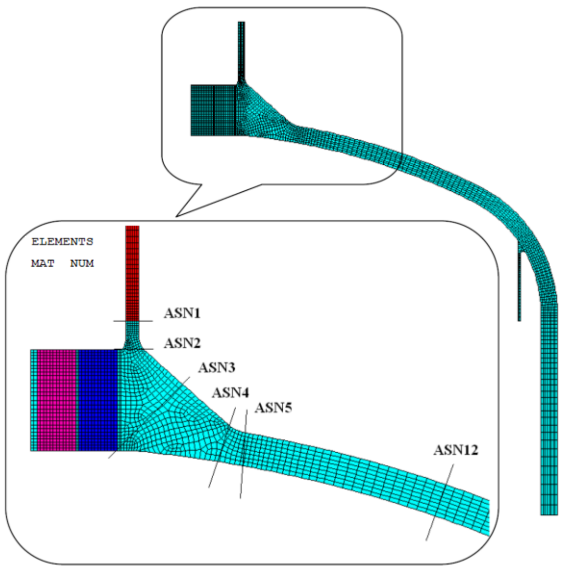
$$\dot{W} = \frac{1}{t}\int Fds \tag{2}$$

where t is time. The wear volume and wear depth is obtained in the design life of components. Enough thickness should be ensured by design. Predicted material loss over the design life of the steam generator due to flow-induced vibration and wear is calculated for steam generator components by relating material loss to local flow velocity through theoretical and empirical correlations. These predicted material losses, along with material loss due to corrosion, are considered in later analyses.

According to the load from former analysis, the strength analysis, fatigue analysis, and fracture analysis are performed using the axisymmetric model of components one by one. For example, analysis model and limiting stress locations of channel head (lower head), tubesheet, lower shell assembly and steam nozzle, elliptical head, upper shell, support ring assembly are shown in Fig.4, in which ASN is Analysis Section Number. And the different material types of elements are shown in Fig 4(b) with different colors. The calculation and analysis are performed for design, normal, upset, emergency, faulted and test condition. And primary inlet nozzle, primary outlet nozzle, primary manway, tubes, cone, feedwater nozzle, et. al., are modeled respectively. The stresses due to the pressure loads, thermal loads and the external mechanical loads are obtained. Thermal loads are temperature gradients (distributions) due to the thermal transients in the components and are also obtained using these finite element analyses.



a) Channel Head, Tubesheet, and Lower Shell Assembly



b) Steam Nozzle, Elliptical Head, Upper Shell and Support Ring

Fig.4, Analysis Model and Limiting Stress Locations of some components

The fatigue evaluation is performed per ASME B&PVC III-1 NB-3222.4(e) [2]. The primary plus secondary plus peak stresses (total stresses) are used in determining the cumulated fatigue usage factor for each of the limiting locations. The design fatigue curves of ASME B&PVC III-1 [2] Appendix I is applied.

Non-Ductile failure evaluation is performed per ASME B&PVC III-1 NB-3211(d) [2] and Appendix G to prevent brittle fracture of components. WRCB 175 [7] "PVRC Recommendations on Toughness Requirements for Ferritic Materials" is applied by the guidance of ASME B&PVC III-1 Appendix G [2]. And K_{IC}, the lower bound static initiation critical K_I value, is applied for the evaluation of the postulated flaw size to ensure non-ductile failure does not occur.

4.2 Evaluation of Components

The stress results are evaluated per ASME B&PVC III [2]. The calculated stress intensities, the ASME Code allowable stress limits, the ratios of the stress intensities to the allowable limits, and the calculated fatigue usage factors for each location are calculated. Herein, the results of two typical locations ASN 2 of tubesheet and ASN 4 of elliptical head (upper head) are shown in Table 1, where P_m is general primary membrane stress intensity, P_L is local primary membrane stress intensity, P_b is primary bending stress intensity, and Q is secondary stress intensity. The stress category method refers to ASME B&PVC III-1 NB Tables NB-3217-1 [2]. The results demonstrate that the stress combination and evaluation under each service level meet the requirement of the ASME B&PVC III [2].

Further more, the results from the fracture mechanics assessment are obtained. The results of the most rigorous transient of typical locations ASN 4 of tubesheet are shown in Table 2, where N19 is primary to secondary side leak test, and T01 is primary side hydro test. The lowest temperature is chose for evaluation. In primary side hydro test, the method of WRCB 175 [7] is applied, and the critical flaw size 17.78 mm is obtained which are detectable by current inspection techniques and are therefore acceptable.

V. Main Factors Affecting Safety

The safety and function of SG depend on the structure integrity of the equipment. The design basis of the Steam Generator is 60 years with 90 percent availability over the design life. The main factors affecting structure integrity and safe operation of SGs are summarized and proposed with the weak parts of SGs. Particularly, more than ten thousands of tubes, which are the key and weak components and are located between the primary and secondary coolant loops of NPP, are susceptible to flow induced vibration, wear, corrosion and seismic damage. The main factors affecting structure integrity and safety of SG are as follows:

1) Earthquake. SG should keep the structure integrity under Safe Shutdown Earthquake. The engineering and society pay more attention on the seismic ability of equipment, especially in post-Fukushima ages. The stress of tubes under SSE is not small, which occupies about 50% of ASME allowable stress (stress limit). The parameter sensitivities of seismic analysis results are studied, such as the effect of another SG of RCS, support, anti-vibration bars (AVBs), and so on. The results indicate

TABLE 1, STRESS ANALYSIS RESULTS AND EVALUATION OF STEAM GENERATOR'S SOME TYPICAL LOCATIONS

Loading Condition	Stress Category	Tubesheet (ASN 2)			Elliptical Head (ASN 5)		
		Stress (MPa)	Allow (MPa)	Ratio	Stress (MPa)	Allow (MPa)	Ratio
Design	P_m	22.16	207	0.11	/	/	/
	P_L	/	/	/	189.19	310	0.61
	P_m+P_b	222.98	310	0.72	/	/	/
	Triaxial σ	28.39	827	0.03	380.96	827	0.46
Level B (Upset)	P_m	25.22	228	0.11	/	/	/
	P_L	/	/	/	198.08	341	0.58
	P_m+P_b	279.46	341	0.82	/	/	/
	Triaxial σ	33.62	827	0.04	398.30	827	0.48
Level A, B & Test (Normal, Upset & Test)	P_m+P_b+Q	479.03	740	0.65	546.45	621	0.88
	Fatigue Usage	/	1	0.23	/	1	0.84
Level C (Emergency)	P_m	26.83	335	0.08	/	/	/
	P_L	/	/	/	200.18	564	0.35
	P_m+P_b	321.51	557	0.58	/	/	/
	Triaxial σ	15.49	827	0.02	410.12	993	0.41
Level D (Faulted)	P_m	28.86	434	0.07	/	/	/
	P_L	/	/	/	242.80	652	0.37
	P_m+P_b	344.26	652	0.53	/	/	/
Test	P_m	33.84	373	0.09	/	/	/
	P_L	/	/	/	227.62	559	0.41
	P_m+P_b	423.93	560	0.76	/	/	/
	Triaxial σ	40.02	827	0.05	448.47	827	0.54

TABLE 2, NON-DUCTILE FAILURE EVALUATION RESULTS OF A TUBESHEET'S LOCATION

ASN Number	Section Thickness T(mm)	Transient No.	Transient Temperature (℃)	K_I	K_{IC}	K_I/K_{IC}	Postulated Flaw Size a (mm)
ASN 4	290.87	N19(Normal)	8	89.76	111.23	0.81	T/4
		U05-B(Upset)	215.01	155.27	219.78	0.71	T/4
		E06(Emergency)	170.33	178.99	219.78	0.81	T/4
		F02(Faulted)	127.50	157.17	219.78	0.72	T/4
		T01(Test)	8	-	-	0.99	17.78

that seismic results are sensitive to support stiffness and AVBs setting number.

2) Fatigue. The equipments of the third generation NPP with 60 years design life endures more cycles of transients than the equipments of the second generation NPP with 30~40 years design life. The fatigue results or equipments' life depends on cycles of transients, stress of components and fatigue parameters of material mainly. Generally, the cumulated fatigue usage factor is large in high stress region and structural discontinuous region. Especially, life reduction of metal components due to the effects of the light-water reactor environment is the research focus in the world [8]. And fatigue monitoring system in service are being researched and developed. Moreover, if the analysis results indicate that the fatigue results don't meet the requirement of ASME code, there are some methods that can be tried to solve the problem, such as improvement of component structure, application of more accurate transients data, application of more accurate analysis method, and so on.

3) Fracture. Fracture mechanics is a main branch of modern mechanics, which studies the fracture strength of the material or component with defects to deal with the catastrophic low stress brittle fracture problem in many actual projects. A series of failures of several large structures occurred, including pressure vessels, storage tanks, ships, gas pipe lines, bridges, dams and many welded parts. There is relationship between reference nil

ductility temperature, fracture toughness, critical defect size and structural integrity. Particularly, the environmental temperature of various conditions affects the fracture analysis results significantly. If the nonconformance case about fracture occurs, some method can be applied to solve the problem, such as modification of service temperature, application of more accurate analysis method, and improvement of nondestructive examination capabilities. The actuality, which is worth noting, is that American Electric Power Research Institute (EPRI) knows its nondestructive examination capabilities, but Chinese companies do not know its NDE capabilities exactly.

4) Flow induced vibration. There are some flow induced vibration accidents of exchanger occurring. For example, after the replacement steam generators had been operating for approximately 11 months, San Onofre Nuclear Generating Station (SONGS) Unit 3 in American Southern California was brought into an unplanned shutdown due to primary to secondary leakage on January 31, 2012. NRC's preliminary analysis conclusion [9] was that the replacement steam generators were damaged by in-plane fluid-elastic instability (FEI) which is the first occurrence within the industry. Flow induced vibration may be relative to steam quality and flow velocity, natural frequency of structure, gap between tubes and AVBs, layout of AVBs, effective AVB contact force, and so on.

5) Fretting wear. EPRI’s data showed that over 50% of PWR (Pressurized Water Reactor) world wide had reported some occurrence of Steam Generation (SG) tube fretting wear, which mainly resulted from flow induced vibration. Fretting wear of steam generator tube generally occurs between tube and anti-vibration bar (AVB) [10]. The fretting wear of tubes in the life time depends on the flow induced vibration response and the wear performance parameter of materials. About 50% new NPPs are being built in China. A lot of material and components are made in China for the first time. Fretting experiment is important and essential for domestic material and components, which is foundation and input of the analysis, prediction and evaluation of fretting failure.

6) Others. Operating experience shows there are some other types of tube degradation occurring in the SG tubes, such as corrosion. There are more than twelve thousands of tubes in one SG, which are the key and weak components and are located between the primary and secondary coolant loops of NPP. They may be susceptible to flow induced vibration, wear, corrosion, seismic damage. If there is a small quantity of tubes being damaged, the method of mechanical plugging can be applied for SGs.

VI. Conclusions

The stress analysis of NPP’s Steam Generator under various loads is performed by finite element methods using the ANSYS Computer Code, such as seismic analysis, fatigue analysis, fracture analysis, flow-induced vibration (FIV) analysis, wear analysis, and so on. The results demonstrate that the stress combination and evaluation under each service level meet the requirement of the ASME B&PVC III. In addition, the main factors affecting structure integrity and safe operation of SGs are summarized and proposed. These comments are expected to be significant for future analysis and design to ensure the safety and function of equipment and NPP. And some methods can be applied to deal with the problem of Non Conformity Report (NCR) that occurs in fabrication of equipments. Some information should be paid more attention on during the operation period of NPP.

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