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IMPROVEMENT IN POST-ACCIDENT INSTRUMENTATION FOR SPANISH NUCLEAR POWER PLANTS

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1. INTRODUCTION AND SUMMARY

Concerns about the adequacy of currently available instrumentation to withstand plant accident conditions and be available during the recovery from an accident raised in 1979 after the accident at Three Mile Island focused attention on post accident instrumentation and the emergency response capabilities. As result the US. Nuclear Regulatory Commission issued Regulatory Guide 1.97 Rev. 2, and the current issue, Rev. 3, that was published in May 1983. In addition was issued NUREG-0737 which contains requirements of the US. Nuclear Regulatory Commission related not only to instrumentation, but also other aspects of emergency response including emergency procedures, technical support center, Safety Parameter Display System, etc.

There are nine operating nuclear power plants in Spain eight of them are US. basic design (Westinghouse and General Electric), as a general rule, the safety technical requirement of the country which the basic design came from is considered applicable in Spain, in particular, Regulatory Guide 1.97 and NUREG-0737 apply to these plants and KTA-3502 for one Siemens-KWU design plant.

This regulation identifies and classified more than 100 variables to be monitored and specify the basic design and qualification requirements for each category. The impact for each plant of this regulation depend of the its generation. For new plants, (third generation) which operating license was issued (1987/1988) later that current regulation, the original design incorporate all requirements for post accident instrumentation. In the case of the second generation of pants (those which operating licenses were issued in the period 1981-1984) current requirements were not included in its design basis. The Spanish Regulatory Body (C.S.N.) required to analyze discrepancies and provide a program to install new instrumentation or to modify the in-place one in order to meet this regulatory guide. For old plants, (first generation) which operating licenses were issued in 1968 and 1970, the design basis are far from current regulation, and a compromise must necessarily be made between the achievable safety benefits and the constraints on installing new systems of instrumentation in old plant and this modifications would be made as part of a general plant safety upgrade program.

This paper is focused in how the regulation issued as consequence of TMI-2 accident, has affected to the Spanish plant describing the main improvement that has been implemented in each plant and providing a general overview of the approach adopted and current status for the different generation of nuclear plants.

2. DISCUSSION OF POST-ACCIDENT INSTRUMENTATION REQUIREMENTS

Regulatory Guide 1.97 defines the instrumentation needed to provide information in case of accident in five types. The types are:

Type A: Are those variables that provide primary information needed to allow operators to take specified manual controlled actions for which no automatic control is provided.

Type B: Are those variable that provide information to indicate whether plant safety functions are being accomplished. This safety function are:
- Reactivity control
- Core Cooling
- Maintaining Reactor Coolant System Integrity
- Maintaining Containment Integrity

Type C: Are those variables that provide information to indicate the potential for breach of barriers to fission product release.

Type D: Are those variables that provide information to indicate the operation of individual safety systems.

Type E: Are those variables to be monitored as required for use in determining the magnitude of radioactive material releases and continuously assessing such releases.

The design criteria are separate in three categories that provided a graded approach to requirements depending on the importance to safety of specific variable. Category 1 provides the most stringent requirements and is specified for key variables, in general provides full qualification (environmental and seismic), redundancy and continuous display and onsite power. Category 2 provide less stringent requirements and applies to instrumentation for indicating system operating status, in general, the most important requirement for this instrumentation is the environmental qualification. Category 3 provide requirements that will ensure high-quality commercial-grade equipment and applies and backup and diagnostic instrumentation.

The essential of this regulatory guide is that identify the minimum number of variables to be monitored, the range selection for each variable that would be sufficiently great to keep instruments on scale at all times and the design criteria for this instrumentation. Is important to remark that the environmental qualification requirements for temperature, pressure, humidity and radiation are set up in relation with the Design Bases Accidents and not specific requirement are considered in relation with Severe Accidents.

In relation with pervious review of this Regulatory Guide (Rev.1) it represent important changes in order to provide new instrumentation, increase range, (in the past, some instrumentation ranges have been selected based on the setpoint value for automatic protection) then wide range are required for monitoring degraded conditions associated with an accident and necessary environmental qualification. It is essential that instrumentation so upgraded does not degrade the accuracy required in normal operation.

KTA-3502 identify the post-accident variables and set-up the design requirements, in a similar way than R.G. 1.97 does.

In addition, Supplement 1 to NUREG-0737, Clarification of TMI Action Plan Requirements, Requirements for Emergency Response Capability, establish that a Safety Parameter Display System should provide a concise display of critical plant variables to the control room operators to aid them in rapidly and reliably determining the safety status of the plant. Although the SPDS will be operated during normal operations as well as during abnormal condition, the principal purpose and function of the SPDS is to aid the control room personnel during abnormal and emergency conditions in determining the safety status of the plant ad in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core.

The basis for this requirement stems from the lack of centralized display capability in the TMI-2 control room. Control room personnel could not easily develop an overview of plant condition. This system must centralize all the variable for critical safety function in order to facilitate the comparison of variables or the integration of various symptoms within the same time frame. Also it could avoid some behaviors such as operator fixation on a limited set of plant anomalies while safety functions where in jeopardy.

3. IMPLEMENTATION APPROACH AND STATUS PER SPECIFIC PLANT

The general approach was to compare the existing instrumentation with Regulatory Guide 1.97 by the licensee and submit a report to the CSN that provides the following information for each variable:

- Instrument range
- Environmental Qualification
- Seismic qualification
- Quality assurance
- Redundancy
- Power supply
- Display

The submittal should identify deviation from the regulatory guide and provide supporting justification or alternatives such as installation new instruments or upgrade existing instrumentation. The submittal should inform about schedule of implementation.

The application of this regulation depend of the date of design and construction for each plant. For an operating plant , is clearly difficult to implement such a large system, and it require flexibility on analyses justification of each deviation, takes into account the benefits obtained versus the impact to the plant in a general point of view.

Under this point of view, we can classify the Spanish NPP in three generation:

3.1 First Generation

The first generation are two plants which operating licenses were issued before that current regulation and the original design is far from this requirement. Different approach have been taken in each case.

- **José Cabrera NPP :** Is a PWR, Westinghouse NSSS design, 160 Mwe, the operating licenses were issued in 1968.

As part of a general plant backfitting in safety systems, during a large shut down in 1985, it was adopted a different approach to the guidelines of R.G. 1.97. This approach consists in the selection of the minimal instrument necessary to provide aid to the operator, in combination with the emergency procedures, in diagnosis and recovery from an accident. Providing the operator with the capability to perform the manual actions specified in such procedures. Then a complete new Post-Accident Monitoring System was implemented with 21 variables to allow monitoring of the plant's Critical Safety Functions:

Critical Safety Function		Variable
SUBCRITICALITY	Key:	None
	Backup:	Wide Range T. cold
		Core Exit Temperature
		RCS Boron Concentration
CORE COOLING	Key:	Core Exit Temperature
	Backup:	Wide Range T. cold
		RCS Wide Range Pressure
HEAT SINK	Key:	S.G. Wide Range Level
		S.G. Narrow Range Level
		A.F.W. Flow
		Steam Line Pressure
		Core Exit Temperature
	Backup:	None
RCS INTEGRITY	Key:	RCS Wide Range Pressure
		Wide Range T. cold
	Backup:	Containment Pressure

		Containment Radiation Secondary System Radiation
CONTAINMENT	Key:	Containment Pressure Containment Radiation Containment Sump Water Level
	Backup:	None
RCS INVENTORY	Key:	Pressurizer Level
	Backup:	Containment Sump Water Level S.G. Wide Range Level

<u>Variables that provide information of operability of Safety Systems</u>

- Containment Sump Water Temperature
- RWST Level
- SI Downcomer Flow
- SI Pump Suction Pressure
- SI Jet Pump Motive Flow
- DWT Level
- Spent Fuel Pit Level

The general functional requirements for these variables were: Redundancy (2 channels per variable); Electrical and physical separation; Powered from separated trains; Seismic and Environmental Qualification (DBA);

As second approach, in 1988 , CSN required a general review of the post accident instrumentation versus R.G. 1.97, identify deviation from the regulatory guide and provide supporting justification or others alternatives. As consequence new or upgrade instrumentation have been identified and installed during the following refueling outages (1990-1994).

- Neutron Flux (1996 refueling outage)
- Degrees of Subcooling
- Air Containment Cooling flow (one channel)
- RHRS flow and temperature (sensor Qualification)
- Containment Hydrogen Concentration (one channel)
- Containment Area Radiation (High Range)
- Containment Isolation valves Position (Qualification)
- Steam Flow (Qualification)
- Vent From SG Safety Relief Valves, Novel Gases and Vent Flow
- Chimney vent discharging

Main exception, accepted by CSN, versus a more modern Westinghouse plant are:

- Reactor Vessel Water Level not provided

- Redundancy for most variables is addressed by providing two channels per variable To strictly address information ambiguity a Third Channel for each variable would be required.
- Same variables such Containment Hydrogen Concentration only one channel is provided.

The degree of implementation is at this time is almost complete , only the Nuclear Instrumentation System remain to be upgraded that is schedule for the next refueling outage. Other important project to be implemented is a Safety Parameter System, now in phase of design.

- GAROÑA NPP :

Is a BWR-3, Mark-I , NSSS design by GE, 460 Mwe, the operating license was issued in 1970.

CSN required to this pant a initial review of plant instrumentation versus R.G: 1.97 Rev. 2 (1982). Even deviations were important but not so large that José Cabrera NPP. Different approach was choose in this case, to upgrade existing instrumentation instead to implement new one.

A general backfitting (Systematic Evaluation Program) was made in the period 1984 - 1987. Improvement in instrumentation was made during this period in relation with Reactor Protection System. In addition a Safety Parameter Display System was installed.

In 1990 CSN required a systematic review of the post accident instrumentation versus R.G. 1.97 and correct or justify each deviation. As consequence important improvement have been made during the following refueling outage. Such as:

- Reactor Vessel Water Level (range, qualification, recorder)
- Temperature Drywell (range , environmental qualification)
- Reactor Pressure (new)
- Temperature of Suppression Pool (new)
- Water Level in the Suppression Pool (physical separation)
- LPCI Temperature exit of cooler
- Temperature SW/LPCI exit of cooler
- Water flow SW/LPCI exit of cooler
- Containment Hydrogen Concentration
- Containment Oxygen Concentration

The degree of implementation is practically total except for the Containment Hydrogen and Oxygen concentration that is scheduled to install in the next refueling outage in 1996.

3.2 <u>Second Generation</u>

The second generation are five plants which operating license was issued in the period 1980- 1985 :

- Almaraz I : PWR, NSSS (Westinghouse design), 930 Mwe, Op. License 1980.
- Almaraz II: PWR, NSSS (Westinghouse design), 930 Mwe. Op. License 1983
- Asco I: PWR, NSSS (Westinghouse design), 930 Mwe. Op. License 1982
- Asco II: PWR, NSSS (Westinghouse design), 930 Mwe. Op. License 1985
- Cofemtes: BWR, NSSS (G.E. design), 994 Mwe, Op. License 1984

Because of current regulation was issued during the final of construction of this plant, this new requirements are not included in its original design basis. However, CSN required to meet with this regulation and upgrade Post- Accident Instrumentation .

The approach for these cases was more strict than old plants and very few exception were accepted.

- ALMARAZ I & II ; ASCO I & II

The four PWR Westinghouse pants are very similar and NSSS instrumentation is practically the identical. Existing qualify instrumentation (narrow range) for the Reactor Protection System is used as well as post-accident instrumentation. More relevant deviations (key variables) are in relation with:

- Wide range requirements not provided
    * RCS Hot Leg Temperature
    * RCS Cold Leg Temperature
    * Containment Sump Water Level
    * Containment Pressure
    * Containment Radiation
    * Pressurizer Level

- Environmental Qualification
    * S.G. Water Level (wide range)
    * Neutron Flux
    * Containment Atmosphere Temperature
    * Core Exit Temperature
    * Containment Sump Water Level (narrow range)

- Requirement of new instrumentation
    * Reactor Vessel Water Level
    * Degrees of Subcooling
    * Containment Hydrogen Concentration
    * Containment Sump Water Temperature
    * Containment Heat Removal
    * Primary System Safety Relief Valve Position

All these instrumentation have been upgraded to meet 1.97 Rev. 3 requirement. However, in relation with the Reactor Vessel Water Level instrumentation the utilities opposed to provide it. The utilities position was based in the high cost and relative benefits of this instrumentation which reliability and operability was in doubt and could mislead to operators and other existing alternative instrumentation (Core Exit Temperature and Degrees of Subcooling) could make its function. CSN disagree with this position considering it is a essential variable as was demonstrated in the TMI accident and the existence of commercial available instrumentation that provide warranty of unambiguous indication.

Plants have changed obsolete process computer and provided Post-accident aids (SPDS) and other functions in the new one.

- COFERNTES NPP

During the final of construction most of the changes necessary were made in order to meet RG 1.97 Rev. 3 . Only a few variables with deviation remain after Commissioning:

- Reactor Vessel Water Level (lack of qualification)
- Suppression Pool Water Level (range)
- Containment Hydrogen Concentration (range)

These instrumentation was update in the first refueling outage. In addition a Safety Parameter Display System (ERIS) is operative since 1986.

3.3 <u>Third Generation</u>

The third generation are two plants which operating license was issued in the period 1987- 1988 :

- Vandellos II: PWR, NSSS (Westinghouse design), three loops, 930 Mwe, Op. License 1987.
- Trillo I: PWR, NSSS (Siemens-KWU design), three loop, 1040 Mwe, Op. License 1988.

In this cases the original design bases incorporate all post TMI requirements, and not significant deviation versus RG 1. 97 Rev.3 have been found.

In relation with computerized aids to operators (SPDS) in Vandellos II NPP, the CSN review found deficiencies in time response (Slow displaying changes). SPDS is installed in a obsolete low capacity plant process computer with many function working at the same time. Plant is planing to changes this computer.

Trillo NPP have not incorporated Safety Parameter Display System.

4. GENERAL OVERVIEW

- Spanish current requirement in Post accident instrumentation
  * US Design plants: RG 1.97 Rev. 3; NUREG-0737
  * German plant: KTA- 3502

- Severe Accident (non specific requirement to provide new instrumentation).

- A great improvement and effort have been made in old plants (first generation) in Post-Accident Instrumentation to meet with the current regulation. Key variables have been addressed, however same exception have been accepted by CSN.

- New plants (second and Third generation) provide all 1.97 Rev. 3 requirements.
  Important improvements and effort have been made in the second generation plants.

- SPDS is provided in US design plants (Westinghouse & GE). Except one plant that is in the design phase.

5. REFERENCES

1. US Nuclear Regulatory Commission " Instrumentation for Light-Water-Cooled Nuclear Power Plants and Environs Conditions During and Following an Accident. RG 1.97 Rev. 3 (May 1983)

2. US Nuclear Regulatory Commission " Clarification of TMI Action Plan Requirements, Requirements for Emergency Response Capability, NUREG-0737 Supplement 1, December 1982

3 . Post Accident Instrumentation, KTA-3502