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**Progress Report to U. S. Department of Energy
on
Grant # DOE-FG02-92ER75781**

**Development of Advanced Direct Perception Displays for Nuclear Power
Plants to Enhance Monitoring, Control and Fault Management**

by

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May 21, 1993

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Background

With recent theoretical and empirical research in basic and applied psychology, human factors, and engineering, it is now sufficient to define an integrated approach to the design of advanced displays for present and future nuclear power plants. Traditionally, the conventional displays have shown operators the individual variables on gauges, meters, strip charts, etc. This design approach requires the operators to mentally integrate the separately displayed variables and determine the implications for the plant state. This traditional approach has been known as the single-sensor-single-indicator display design and it places an intolerable amount of mental workload on operators during transients and abnormal conditions.

A new alternative approach is the use of direct perception interfaces. Direct perception interfaces display the underlying physical and system constraints of the situation in a directly perceptual way, such that the viewer need not reason about what is seen to identify system states, but can identify the state of the system perceptually. There has been progress using this display design approach. Beltracchi (1987; Moray, Jones, Rasmussen, Lee, Vicente, Brock, and Djemil, 1992) has developed a direct perception display for the steam generating systems of a pressurized water reactor (PWR). This direct perception display is based on the Rankine Cycle, which shows the physics constraints of the underlying thermodynamic processes. In addition, Jens Rasmussen has worked on developing advanced displays for nuclear power plants that also display the system constraints and boundary conditions for operator performance (see Figure 1).

We expect that displays which show the dynamics of fundamental physical laws should better support operator decisions and diagnoses of plant states. The purpose of this research project is to develop a suite of direct perception displays for PWR nuclear power plant operations.

Identification of Variables

For the past several months, we have familiarized ourselves with the general operations and physics of pressurized water reactors (PWR) by reviewing the Technology Manual: Pressurized Water Reactors Westinghouse Design from the US NRC Technical Training Center and the Braidwood License Training Fundamentals manuals.

Figure 1: An illustration of a display for cold start-up of a research reactor from source level to full power. The display changes the start-up task from following a set of rules to navigating the power/period indicator along the boundaries of a space on a map. The display indicates the boundary conditions around the acceptable power transition trajectory (Jens Rasmussen, personal communication, March 9, 1993).

Through correspondence between Dal Vernon Reising and Commonwealth Edison Production Training Center personnel, we obtained a list of most commonly used variables for normal operations (see Table 1). Among these variables, some are directly measured from sensors while others are derived or calculated. The measured variables include: hot leg temperature, cold leg temperature, excore core neutron flux, control rod positions, pressurizer water level, and pressurizer pressure. The derived variables include: T_{ave} , axial offset, quadrant power tilt ratio, power range channel deviation, KW/FT, and DNBR. Calculations and definitions of some of these variables are shown in Appendix 1. All the above variables are commonly calculated and monitored to ensure safe operation.

Identification of Safety Parameters

By reviewing this list of variables and the technical manuals, we have identified the three critical safety parameters for a PWR, namely, the reactor coolant system (RCS) pressure, the departure from nucleate boiling ratio (DNBR), and the power density (the amount of energy generated per foot of fuel in KW/ft).

Departure from nucleate boiling must be avoided to prevent cladding failures, which could lead to possible radiation contamination of the RCS. The technical specification for the minimum DNBR for normal operation is be 1.33. Once the DNBR falls below 1.00, which is the physics constraint for the DNBR, partial film boiling of the RCS coolant will occur along the fuel cladding surface.

The RCS pressure is directly related to DNBR, e.g., a decrease in RCS pressure decreases DNBR. The greater the RCS pressure, the more efficient the heat removal to the secondary system will be. Normal operating constraint for the RCS pressure is 2235 psig. The RCS pressure limit also insures the integrity of the containment barrier to the release of fission products.

The power density limit prevents centerline fuel temperature from exceeding the melting temperature. This safety parameter is expressed as heat energy production per foot of fuel rod since the fuel temperature is not directly measured. If the fuel melting temperature is reached, thermal expansion of fuel pellets may cause cladding damage. The normal operating limit for power density is 12.6 KW/ft (determined from the Commonwealth Edison training manuals for the Braidwood power station).

We have develop preliminary graphics for direct perception displays using these three safety parameters in a manner similar to Rasmussen's example shown in Figure 1.

Table 1. Critical variables for Normal Operation

Variables	Units
Hot leg temp, T_h	F
Cold leg temp, T_c	F
Excore core neutron flux monitors	% power
Turbine load	MW
Pressurizer water level	%
Pressurizer pressure	Psia
Pressurizer relief valve	open/closed
Rod position	steps
Core power rate	%/s
T_{ave}	F
Axial offset, ΔI	%
Quadrant power tilt ratio	no units
Power range channel deviation	%

Figures 2, 3 and 4 show the operating, safety, and engineering constraints for each pair of these three safety parameters.

After discussions with Jens Rasmussen, the clean cold start-up scenario was chosen as a starting point for this project. The NRC manual provided a simplified start-up procedure which was used as the basis for the task analysis. Cold start-up steps were divided according to the operational modes defined by the Technical Specifications from cold shutdown (Mode 5) to power operation (Mode 1) (see Appendix 2 for details of the start-up procedures).

It was apparent from the task analysis on the cold start-up procedure that the variables used were not the safety parameters. The task-relevant variables for RCS heat-up, prior to pulling control rods, are the RCS pressure and RCS temperature, which operators control and monitor to determine plant state.

Specifically, there is a limit for the rate at which the RCS may be heated up. The technical specification for the maximum heat-up rate is 100 F/hr. The Braidwood power station has a designated maximum heat-up rate of 50 F/hr. Moreover, there is a close relation between RCS temperature and RCS pressure. Each need to be increased in a way that, at given pressure, the temperature stays below the saturation temperature. Currently we have generated a temperature vs. pressure curve for the RCS heat-up from Mode 5 to Mode 3 (see Figure 5). This figure shows the end states of both temperature and pressure at each mode :

Mode 5 --> 4,	Temp. : 160 F --> 200 F	Pressure: 100 psig --> 325 psig
Mode 4 --> 3,	Temp. : 200 F --> 350 F	Pressure: 325 psig --> 425 psig
Mode 3 --> 1,	Temp. : 350 F --> 547 F	Pressure: 425 psig --> 2235 psig

The graph also illustrates the operating and safety constraints for RCS pressure and RCS temperature.

RCS normal operation pressure = 2235 psig ; tech specs max. pressure = 2735 psig
RCS design temperature = 650 F
Pressurizer design temperature = 680 F

Once the NPP reaches Mode 3, the operators no longer concentrate on RCS temperature and RCS pressure. At this point, the operators' task is to take the reactor to criticality. This is done by pulling control rods and adjusting the boron concentration. In most plants, a critical control rod height is chosen and boron concentration is calculated from that height. However, criticality may not occur exactly at the calculated control rod height, therefore it

DNBR vs. KW/ft. Constraints for a PWR

Figure 2: The DNBR vs KW/ft constraints for a PWR.

DNBR vs. RCS Pressure Constraints for a PWR

Figure 3: The DNBR vs RCS Pressure constraints for a PWR.

KW/ft. vs. RCS Pressure Constraints for a PWR

Figure 4: The KW/ft vs RCS Pressure constraints for a PWR.

System Constraints and Desired Trajectory through the Temperature vs Pressure Space During Cold Start-Up for a PWR

Figure 5: The trajectory through the Temperature vs. RCS Pressure during cold start-up for a PWR.

is important to have an interface that would assist in predicting the range in which criticality could occur.

Some ideas for representing this information are currently being examined. One idea is to use the $1/M$ plot concept which shows when criticality will occur with respect to the number of movement steps of the control rods. $1/M$ is the inverse of neutron count rate which will approach zero as criticality is approached. This is because the count rate increases with an increase in the control rod steps. Theoretically operators can predict criticality, i.e., when $1/M$ reaches zero.

Because the NRC and Braidwood training manuals which we have do not provide us with all the necessary information for the cold start-up procedure, we will acquire the specific parameters and variables used by the operators during cold start-up from the Braidwood facility. We hope that this will then allow us to complete the suite of displays which will support the cold start-up procedure by directly displaying variables and parameters that operators currently have to monitor on separate instruments or determine from reference tables.

Identification of Required Hierarchy Levels

As stated in the proposal, the two most important hierarchies are the "means-ends" hierarchy and the "abstraction" hierarchy. The means-ends hierarchy represents the goals and intentions of the operators while the abstraction hierarchy represents the structural and functional designs of the physical system.

We have attempted to display the abstraction hierarchy's functional level for the primary system of the PWR by using a lattice approach (Moray, 1990) (see Figure 6). The primary goal of the primary system is to transfer heat to the secondary system. This heat is used in the secondary side to generate steam which turns the turbines and thereby generates electricity. Figure 6 shows how heat generation and safety concerns are related functionally.

From this general lattice, we are generating separate lattices, one each for the physics, engineering, and safety constraints relevant to the primary system. We will also produce similar lattices for the secondary system.

Functional Lattice for Primary Side of a PWR

Figure 6: The functional lattice for the primary system of a PWR

Development of Interface Graphics

Having generated static graphics for system variables and safety parameters, we have also begun looking into implementing these graphics into real-time animated displays. Each of the SUN workstations is set up and different options for graphic software are being discussed with personnel in the National Center for Supercomputing Applications (NCSA) and the Computing and Communications Services Office (CCSO). We will be programming the displays in C++ and will be obtaining real-time simulation data from the Braidwood training facility with which to drive the direct perception displays that will be created.

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Appendix 1. Definitions and Calculations

1). DNBR (Departure from Nucleate Boiling Ratio)

$$\text{DNBR} = \frac{\text{heat flux required for DNB to occur}}{\text{actual local heat flux}}$$

2). ΔI (Axial Flux Difference)

--is the difference in normalized flux signals between the top and bottom halves of a two section excore core neutron detectors; measures the axial power symmetry.

ΔI is positive, more power at the top of the core

ΔI is negative, more power at the bottom of the core

3). HCF (Hot Channel Factors or Peaking Factors)

$$\text{HCF} = \frac{\text{peak core heat output}}{\text{ave. core heat output}} = \frac{\text{max. neutron flux}}{\text{ave. neutron flux}}$$

4). Average Linear Power Density

$$\text{KW/FT} = \frac{\text{core license power} * 0.974}{\text{total fuel rod length}}$$

$$5). T_{\text{ave}} = (T_h + T_c) / 2$$

Appendix 2. Brief Description of the Cold Start-up Procedures

Mode 5 (cold shutdown) to Mode 4 (hot shutdown)

- verification and permission prior to heat up
- increase RCS pressure by heating up the pressurizer at a rate of 50 F/hr (Braidwood specifics; Tech Spec allows up to 100 F/hr).
- establish supporting systems (steam generators, main steam isolation valves, condenser vacuum)
- heat up pressurizer to 430 F, RCS pressure ~325 psig
- start Reactor coolant pump
- check water chemistry when RCS Temp = 200 F
- heat up RCS to Temp = 350 F and Pressure = 325 psig

Mode 4 (hot shutdown) to Mode 3 (hot standby)

- check ECCS
- before RCS pressure reaches 1000 psig, open cold leg accumulator isolation valve
- verify steam line safety injection logic auto reset when Tave>540 F
- steam dump control set to keep RCS temp = 547 F
- RCS Temp = 540 - 547 F and pressure = 2235 psig

Mode 3 (hot standby) to Mode 1 (power operation)

- calculate shutdown margins
- withdraw control rods
- @ 10^{-8} amps, record data (e.g., rod positions, power, boron concentration, Tave)
- @ 1 % power, start feedwater pumps
- @ 8 % power, start secondary systems
- @ 15 % power and Tave = Tref, set reactor control system to auto
- continue power increase till full power

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